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Interim Source Term Assumptions for Emergency Planning and Equipment Qualification

ORIGINAL

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Prepared for
U.S. Nuclear Regulatory
Commission

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FOREWORD
BY
NUCLEAR REGULATORY COMMISSION STAFF

The Nuclear Regulatory Commission (NRC) staff is in the process of reappraising its regulatory position relative to its recommendations on the assumptions to be used in calculating light water reactor accident source terms. Two initial steps in the process were the development of "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions" (NUREG-0771 for Comment), and "Technical Bases for Estimating Fission Product Behavior during LWR Accidents" (NUREG-0772). To aid the staff in evaluating the information contained in NUREG-0772 and the comments on NUREG-0771, a contract was awarded to the Oak Ridge National Laboratory's (ORNL's) Health and Safety Research Division. Included in the contract were tasks related to a review of the technical bases and models presented in NUREG-0772, review of current status of on-going research, and recommendations regarding mechanistic assumptions and models that the staff might use in developing criteria for equipment qualification and emergency preparedness purposes. Some of the ORNL work reported here is to be used in the staff assessment leading to the final publication of NUREG-0771 in July 1982.

Although the authors conclude that the technical bases of NUREG-0772 and the current (early 1982) status of severe accident source term research yield some changes in specific radionuclide group releases to the atmosphere, the overall results are judged by the staff to be not significantly different from the source terms contained in the Reactor Safety Study (WASH-1400). The report also highlights that substantial uncertainties in the models remain and recommends further work to resolve the uncertain issues.

A draft of the report was reviewed by a group of peer reviewers. A meeting of the contractor, the peer reviewers, and the Nuclear Regulatory Commission staff was held on March 18, 1982 to receive and discuss the peer comments. The comments received at the peer group meeting are discussed by the contractor in appendix B of this report. Some of the peer reviewers did not receive the draft report in time to complete their reviews and to present their comments at the peer group meeting. Their comments were recieved subsequent to the peer group meeting and, due to the expiration of this contract, were not responded to by the contractor. All of the comments received have, however, been included in appendix C of this report. Several peer reviewer's comments indicate substantial disagreements with the calculations of source terms presented in the report.

The findings of this report and the peer group comments indicate to us that the technical bases developed in the NUREG-0772 and the current status of the ongoing source term research are inadequate to make immediate, across the board, changes to accident source term guidance. The final version of NUREG-0771 in July 1982 is expected to indicate findings that may lead to some limited changes, however. The ongoing programs of the Nuclear Regulatory Commision's Office of Research are expected to provide improved technical bases and models for more significant changes at a later date. The details of NRC's research related to severe accident source terms are given in "Nuclear Plant Severe Accident Research Plan"(NUREG-0900 - Draft). The research plan projects the development of updated fission product source term in FY 1983 to be followed by a long term verification program over the period FY 1984 - 1988.

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SUMMARY

The source terms recommended in the current regulatory guidance for many considerations of light water reactor (LWR) accidents were developed a number of years ago when understandings of many of the phenomena pertinent to source term estimation were relatively primitive. The purpose of the work presented here was to develop more realistic source term assumptions which could be used for regulatory purposes for two specific considerations, namely, equipment qualification and emergency planning. More specifically, source term assumptions were to be developed for two accident spectra, one for equipment qualification and one for emergency planning, so that updated source term perspectives could be factored into the interim guidance considerations for those two areas. For equipment qualification, the spectrum was to be such that an improved description of the accidents included in the design basis for LWR's could be obtained. For emergency planning, the spectrum was to be such that an updated description of the accidents required for probabilistic risk assessments, or at least for consideration of a wide range of accidents of all possible severities, could be obtained.

The emphasis of this work was on developing appropriate assumptions for estimating the magnitudes of the radionuclide releases for various groups of accidents in each of the accident spectra of concern. The overall approach taken was to adopt basic assumptions and models previously proposed for various aspects of source term estimation and to modify those assumptions and models to reflect recently gained insights into, and data describing, the release and transport of radionuclides during and after light water reactor accidents. Many of the assumptions and/or procedures adopted here were based on work presented or summarized in the recently published Technical Bases Report (USNRC, 1981). In particular, the method proposed in that report for estimating the initial radionuclide releases from the core materials was adopted. In addition, the insights provided by that document concerning radionuclide retention in the primary coolant system were used to develop assumptions for estimating escape from the coolant system. Furthermore, previously proposed procedures summarized in that report were used to develop assumptions for estimating escape fractions from the containment; namely, models and procedures developed during and shortly after the Reactor Safety Study (RSS; USNRC, 1975) for considering transport of radionuclides within the containment were adopted.

The adopted assumptions and procedures were illustrated by developing representative estimates of source terms both for emergency planning and for equipment qualification. For some accident sequences previously estimated to dominate emergency planning considerations, in particular, for certain transient-initiated accidents and some small-break loss-of-coolant accidents, some of the radionuclide releases were predicted to be noticeably smaller than previously anticipated because of much larger radionuclide retention in the PCS. In contrast, some of the radionuclide releases to the environment for certain other dominant sequences

were estimated to be larger than formerly assumed for two reasons: larger estimates of the releases from the core materials and later releases of some radionuclides from the core materials and/or coolant system. Source terms were estimated to be smaller for many of the less important emergency planning sequences because natural processes were predicted to be much more effective in preventing releases than had been previously assumed.

For equipment qualification, for sequences involving partial melting of the core, the estimates of the releases of some of the more volatile species to the containment were much larger than those currently in use but the estimates of the releases of some of the less volatile species were smaller than those currently recommended. As was the case for emergency planning, the larger releases to the containment were the result of larger, later releases from the core materials, whereas the smaller releases to the containment were the result of smaller initial releases. For all types of accidents, the possible range of initial distributions of the radionuclides between the containment atmosphere and water in the containment was predicted to be much larger than that previously assumed. The broader range was the result of more realistic consideration of the possibilities.

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U.S. Nuclear Regulatory Commission, 1975. Reactor Safety Study. WASH-1400 (NUREG-75/014), USNRC, Washington, D.C.

U.S. Nuclear Regulatory Commission, 1981. *Technical Bases for Estimating Fission Product Behavior During LWR Accidents,* NUREG-0772, USNRC, Washington, D.C.

1. INTRODUCTION

1.1 PERSPECTIVE

1.1.1 Purpose

The purpose of the work reported in this document was to review some assumptions currently used for regulatory guidance related to light water reactors (LWR's). In particular, it was to consider the appropriateness of certain assumptions concerning post-accident radionuclide releases and behavior in the containment which are used for two distinct purposes:

1. equipment qualification, and
2. emergency planning.

Based on the review, the project was to provide recommendations for updated source term assumptions for use in each of those two areas. More exactly, the project was to describe the accident spectrum designated to be appropriate for considering each of the two indicated areas and it was to develop assumptions for estimating the magnitudes of the radionuclide releases for accidents within each of those spectra.

This document presents the assumptions developed during the project reported here. A related report (ORNL/TM-8275) presents the detailed technical considerations which factored into the development of those assumptions.

1.1.2 Equipment Qualification Versus Emergency Planning

In equipment qualification, the immediate concern is the potential hazard to the equipment inside the nuclear plant during and after an accident. If a not-too-serious accident occurs, the necessary equipment to keep the accident from progressing should be able to operate. If a more serious accident occurs, the appropriate equipment to monitor the conditions and to ensure prolonged confinement of the radioactivity within the containment should continue to function. Equipment qualification includes estimation of the potential radiation doses to the equipment-of-concern.

In emergency planning, the primary concern is the potential radiation hazard to the human population located in the vicinity of the nuclear plant at the time of an accident. If a serious accident occurs, certain members of that population might need to take some type of protective action to mitigate the possible radiation doses. Emergency planning includes estimation of the potential radiation doses to the human population, along with development of procedures which might be employed to implement various types of protective actions.

Obviously, the physical domains of interest for equipment qualification and for emergency planning are essentially nonoverlapping. Whereas the ultimate region of interest for the former purpose is inside the containment (or reactor building), the region of interest for the latter purpose is outside the containment (nuclear plant).

Current regulatory guidance for source terms for equipment qualification is based on the concept of the Design Basis Accident (DBA). In contrast, current considerations for emergency planning are often based on a risk assessment approach. A comparison of the design basis approach and the risk assessment approach is presented in Table 1.1. [For a brief review of the history of regulatory practice in these two areas, the reader might consult Section C.3 of Appendix C of *Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions* (Pasedag, Blond, and Jankowski, 1981). For descriptions of the source terms currently used in both these areas, the reader might refer to Appendix E of the other report for this project (Niemczyk and McDowell, 1982).]

1.1.3 Accident Spectrum Approach

In this report, the assumptions in the current regulatory approaches for equipment qualification and for emergency planning are updated by "mechanistic" consideration of a spectrum of accidents appropriate for each area. In this mechanistic treatment, a plausible scenario is developed and utilized for each accident investigated. Such consideration of individual accidents leads to an improved description of the overall spectrum for each area. For equipment qualification and its associated design basis approach, such "realistic" consideration of the indicated spectrum leads to a more meaningful description of the design basis. For emergency planning and its corresponding risk assessment approach, such realistic consideration of a broad spectrum leads to an updated basis for estimating the risk from both individual accidents and all accidents.

1.1.3.1 Equipment Qualification

1.1.3.1.1 Current Approach

Many current regulatory procedures, such as those for equipment qualification, are based on the concept of defense-in-depth against what were perceived some years ago to be the worst credible nuclear reactor accidents. The defense-in-depth approach includes both diversity and redundancy of safety functions and systems, along with multiple physical barriers to prevent releases of radioactivity to the environment. At the time the current regulations were formulated, the worst credible accidents were taken to be enveloped by the fictitious Design Basis Accidents (DBA's). In particular, the worst credible accidents, or the DBA's, were taken to include the following possibilities:

Table 1.1 Differences between the DBA approach and the probabilistic risk assessment approach

	DBA approach	Risk assessment approach
Accidents	DBA's only	Degraded core and meltdown accidents
Descriptions	Nonmechanistic[a] "Conservative"	Mechanistic "Realistic"
Framework	Deterministic (more severe accidents than DBA's are not credible; make the world absolutely safe from DBA's)	Probabilistic (all accidents are included in a weighted fashion according to their probabilities of occurrence; consider overall acceptable risk/time)

[a] The current regulatory guidance for DBA's is based on a nonmechanistic, conservative approach. In contrast, the approach for DBA's in this report is a "mechanistic," "realistic" one.

1. loss-of-coolant accidents (LOCA's);
2. rod ejection accidents for pressurized water reactors (PWR's) and rod drop accidents for boiling water reactors (BWR's); and
3. steam generator-tube ruptures and steamline breaks outside the containment for PWR's.

All of these accidents were assumed to result in less than complete meltdown of the reactor core. (See Section A.2 in Appendix A for general descriptions of each of these types of accidents.)

The postulated releases of radioactive materials for the DBA's were not developed on a mechanistic basis. Instead, the amounts of the radioactive materials released as the result of any accident included in the DBA envelope were taken to be large enough that they were thought to bound the amounts of materials which could be released in any credible accident. The population-of-concern for such accidents was assumed to be the human population. Consequently, to compensate for those radionuclides which would be released in any DBA accident in presumably small but unknown quantities, extra amounts of very radiologically effective radionuclides such as the radioiodines* were added to the anticipated DBA releases. Thus the postulated values of those source terms were thought to include a conservative margin. In no sense were the postulated releases given in the regulatory guidance assumed to be realistic.

Unfortunately, the releases postulated on consideration of the hazard to the human population were adopted for considering the hazard to other populations-at-risk. For example, the DBA source terms initially proposed for siting guidance were adopted for equipment qualification guidance.

The DBA used for both those purposes is the DBA-LOCA and involves the largest assumed releases of radionuclides of any of the DBA's. The characteristics for the DBA-LOCA are determined from the following assumptions:

1. substantial melting of the reactor core occurs, implying less than optimal performance of those systems designed to prevent melting;
2. containment integrity is maintained; and
3. the engineered safety features (ESF's) designed to mitigate the consequences function properly.

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*Some of the radioisotopes such as some of the radioiodines result in a much larger radiation dose per unit of activity (e.g., μCi) than other isotopes.

1.1.3.1.2 TID-14844

In current regulatory guidance for equipment qualification, the magnitudes of the radioactive releases after any potential accident within the design basis are taken to be encompassed by the releases suggested in TID-14844 (USAEC, 1962). The releases constituting the TID source term were taken to result from the DBA-LOCA, that is, from complete rupturing of a major coolant system pipe followed by a partial melting of the core and a release to the containment of some part of the radionuclide inventory of the core. Possible scenarios for the accident were not considered. At the time the TID source term was suggested, not enough data existed to enable the radionuclide releases resulting from various accident conditions to be estimated realistically. Although much more information has become available over the years since it was formulated, the TID source term has continued to be part of the regulatory guidance for many purposes such as equipment qualification.

Since the current regulatory bases were established, it has become apparent that the TID source term is not applicable for at least some purposes. In particular, for those areas in which the radiation hazard to the population-of-concern is dominated by radionuclides other than the noble gases and the halogens (the major components of the TID source term), there have been indications that the TID source term might not be conservative.

1.1.3.1.3 Approach in This Report

For equipment qualification, the approach in this report is in part analogous to the current approach in that the design basis concept is utilized.* However, the approach taken here is to generate source terms using "realistic" analyses of possible worst-case accidents rather than using bounding (enveloping) analyses of undefined accidents. This approach is adopted because the consequences of some of the accidents investigated here are potentially large and because what is conservative is not always apparent. The accident analyses undertaken are then used to determine a more reasonably based envelope for the design basis source terms.

1.1.3.2 Emergency Planning

1.1.3.2.1 Current Approach

There is currently no regulatory guidance for emergency planning source terms. Thus, a "modified design basis" source term is used by some. (In particular, the TID source term is used in combination with an assumption of massive breaching of the containment.) In contrast, the current regulatory guidance for environmental impact analyses is followed by others.

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*The restriction of the accidents considered for equipment qualification to just those within the current design basis was part of the definition of the scope of this project by the funding agency.

Current regulatory guidance for environmental impact analyses requires consideration of all possible accidents in the context of their probabilities of occurrence; that is, probabilistic risk assessments are indicated. Not only are lesser accidents, such as those within the design basis envelope, considered, but also much more severe accidents, such as those involving melting of the complete reactor core, are included.

In a probabilistic risk assessment, the consequences of various types of accident sequences representative of the total accident spectrum are estimated. Then, the overall risk for those accidents is calculated by weighting the representative accidents according to both their likelihoods of occurrence and their estimated consequences. Thus the approach attempts to put into perspective the relative importance of low-probability high-consequence accidents and that of high-probability low-consequence accidents, as well as that of accidents with all other possible combinations of probabilities and consequences. The starting point for such an assessment is the description of the source terms for the indicated accident sequences.

When the original licensing basis was being developed, little information was available about the likelihoods of occurrence of various types of possible accidents. In addition, as was already indicated, little information was available about the magnitudes of the associated potential releases of radioactivity. Thus a risk assessment approach was not reasonable. However, since the development of the original licensing basis, much more information has become available on both the probabilities of the various types of accidents which might occur and the amounts of radioactive materials which might be released as a result of any of them. Although some of this information has subsequently been factored informally into certain considerations of emergency planning, none of it has become a part of official regulatory guidance for emergency planning. Thus, unlike consideration of equipment qualification, some treatments of emergency planning involve the use of source term assumptions based on some of the more recent information. Unfortunately, some other treatments are still based on the source terms used for equipment qualification.

1.1.3.2.2 Reactor Safety Study and Reactor Safety Study Methodology Applications Program

In some current considerations of emergency planning, the radionuclide releases after any accident are taken to be described by the assumptions used in the relatively recent Reactor Safety Study (RSS; USNRC, 1975). The work performed for that study was the most comprehensive attempt to date to assess the relative contributions to the risk to the human population of various types of meltdown and other degraded core accidents. Because that work had as its goal the estimation of the risk to the human population, its emphasis was on accidents involving complete meltdown of the reactor core, although it also considered accidents involving limited core damage.

In that work, two specific nuclear plants were reviewed in detail. Possible accident sequences covering a large portion of the entire accident spectrum for each plant were delineated. Event trees and fault trees were constructed and estimates of the probabilities of various types of accidents were developed. The postulated accident sequences were grouped into nine PWR categories and five BWR categories, according to those features upon which the potential atmospheric pathway consequences depend. Phenomenological models were used to consider the behavior of the radioactive materials and to describe the transport and movement of the radioactive material both inside and outside the reactor containment for a spectrum of accidents for each plant. In addition, the consequences to the human population were described for representative accidents in each spectrum. Furthermore, the aggregate risk to the population from all the various types of potential accidents was estimated. Thus, the accident sequences dominating the risk to the human population for those two plant designs were identified.

Since the completion of the RSS, the Reactor Safety Study Methodology Application Program (RSSMAP) has applied the methods developed in the RSS to the consideration of the accident spectra for four other plant designs (Carlson et al., 1981; Kolb et al., 1981; Hatch, Cybulskis and Wooton, 1981; Cybulskis, 1981; Wooton, 1981). As in the RSS, both the probabilities and the magnitudes of the radionuclide releases of accident sequences potentially important for estimating the risk to the human population were quantified. Unlike the RSS, the overall risk to the human population was not estimated for any of the RSSMAP plants. To ensure consistency with the RSS, the basic assumptions in the RSSMAP regarding source terms were taken to be the same as those used in the RSS.

At the time the RSS source terms were generated, they were thought to be conservatively realistic, that is, they were thought to be chosen to err on the conservative side when not enough information was available to make a realistic assessment. Recently, however, it has been suggested that the source term assumptions used in both those studies, and therefore in some current considerations of emergency planning, might be highly conservative for a variety of reasons. Namely, recent research has filled in some of the gaps existing in the knowledge at the time the RSS was performed.

1.1.3.2.3 Approach in This Report

The approach in this report is to utilize the risk assessment framework currently used by some for emergency planning. The emphasis here is on updating the assumptions to be used for estimating the magnitudes of the associated radionuclide releases on the basis of relatively recent findings. As is required by either consideration of emergency planning for individual accidents or estimation of the risk from all possible accidents, the analyses are directed to obtain realistic or best estimate values, rather than conservative values, for the emergency planning source terms. Unlike the situation for a DBA analysis, best estimates for the individual accidents are used both for accident planning and for a risk analysis; they are not used to generate an envelope of the source terms for all relevant accidents.

1.1.4 Technical Bases Report

A recent report, entitled *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, was written by experts at several laboratories (Battelle Columbus Laboratories, Oak Ridge National Laboratory, and Sandia National Laboratories) to summarize all the post-RSS work relevant to estimating the magnitudes of the radionuclide releases for LWR accidents (USNRC, 1981). Its views are thought by many to represent the state-of-the-art of a number of non-probabilistic aspects of source term estimation.

The work performed for the Technical Bases Report considered in detail many of the factors affecting the releases of radionuclides from the core materials, the transport of those radionuclides in the primary coolant system, the subsequent transport in the reactor containment, and the eventual escape of some of the radionuclides into the environment. Portions of several hypothetical accident sequences were considered for each of four basic types of nuclear plants. Estimates of the effects of the important factors affecting the magnitudes of the radionuclide source terms were made separately for the initial releases, the transport in the coolant system and the transport in the containment. The interactions between the effects in the various portions of the overall problem were not treated.

1.1.5 Scope

The scope of this study includes all major factors affecting the magnitudes of radionuclide source terms for light water reactor accidents involving core damage. The emphasis is on determination of assumptions and/or procedures which can be used to estimate those magnitudes for certain regulatory purposes.

Some of the assumptions and procedures adopted here for estimating the magnitudes of the source terms for accidents are based on the work in the Technical Bases Report. However, the scope of this project goes farther than that of the Technical Bases Report in that the methods recommended in that study are applied to complete accident sequences rather than to just portions of them. Furthermore, this study quantifies the effects of factors only alluded to in the Technical Bases Report. In particular, it considers interdependencies of effects not previously treated. In addition, it attempts to place the new source term assumptions and illustrative estimates in an appropriate framework.

Some of the other assumptions and procedures adopted here are based on the work in the Reactor Safety Study. The scope of this project is much different than that of the Reactor Safety Study in that the goal here is consideration of only source term magnitudes and not the estimation of the overall risk. In addition, the emphasis here is on obtaining source terms suitable for regulatory purposes. Furthermore, this project is concerned not only with source terms appropriate for considering the hazards to humans but also with source terms appropriate for considering the hazards to equipment.

1.1.6 Scale

Although the scope of the project reported here was rather broad, the scale of the project was very limited. Unfortunately, many aspects of estimating source terms are extremely complicated and have not been adequately treated in the past. Because of the limited scale of this study, it was not possible to consider most topics at a greater level of detail than had been done previously. In addition, it usually was not possible to develop new approaches for problems which had been either inadequately or even inaccurately treated in the past.

1.1.7 Overall Approach

To develop assumptions appropriate for estimating the amounts of radionuclides which might be released during various reactor accidents, previous source term work, as summarized in the Reactor Safety Study and the Technical Bases Report, was reviewed. Based on this review, a set of assumptions and procedures was adopted and/or developed for the estimation of post-accident source term magnitudes. To test the overall approach and to illustrate its use, it was applied to obtain estimates of radionuclide releases for a broad spectrum of accidents.

1.2 PRESENTATION

Chapter 2 of this report describes the processes concerning the release of radionuclides from the core materials and the subsequent transport of those materials within the nuclear plant which are relevant for estimating the consequences of various accident sequences. The emphasis is on presenting a qualitative description of the most important factors affecting the source terms.

Chapter 3 summarizes the assumptions adopted and developed in this study to estimate source terms for a spectrum of light water reactor accident sequences. It discusses both the methods indicated to describe individual accidents and the procedures suggested to place those accidents in an accident spectrum framework.

Chapters 4 and 5 present the source terms which have been estimated in this study for illustrative purposes. In addition, they consider the effects of such new source terms on equipment qualification and emergency planning.

Chapter 6 outlines some of the limitations of the work presented here. In addition, it lists some suggestions for future work.

Appendix A reviews the terminology used throughout the report for discussing reactor accidents. In addition, it includes descriptions of the accident sequences specifically investigated in the study.

This report differs from the other report for this project, ORNL/TM-8275 (Niemczyk and McDowell-Boyer, 1982), in that whereas this report emphasizes just the assumptions which, based on this project, are thought to be most appropriate for source term estimation, the other report emphasizes the rationale used to choose the adopted assumptions from among the many possibilities. The other report also considers in much more detail the factors affecting the use of the adopted assumptions to obtain numerical estimates for source terms. In addition, it describes the details of the illustrative calculations.

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2. QUALITATIVE DESCRIPTIONS OF SOURCE TERMS

2.1 INTRODUCTION

Releases of radionuclides from the core materials could occur as a result of many types of accidents. In lesser accidents, such as limited-core-damage accidents, restricted overheating of at least some portion of the core usually would occur. In more severe accidents, such as meltdown accidents, extensive overheating of almost the entire core would occur.

A limited-core-damage accident typically would result from a basic system failure such as a coolant line break or a transient event (event in which either the reactor power increases too much or the heat removal capacity of the coolant system drops too low). Typically, the requisite engineered safety features (ESF's) would function as anticipated and would soon return the cooling of the core to normal. Although a meltdown accident would usually result from one of the same basic accident initiators (coolant line break or transient) as a limited-core-damage accident, unlike such an accident, it would be accompanied by a failure of one or more of the ESF's required to maintain or restore adequate cooling. Thus the ESF's could not return the reactor to normal conditions and the core would melt "completely." An accident intermediate between a "typical" limited-core-damage accident and a complete meltdown accident, for example an accident such as the one at Three Mile Island, frequently would result from one of the basic accident initiators, accompanied by partially impaired performance of one or more of the requisite ESF's. Consequently, the cooling might not be restored to an adequate level until after severe damage had been done to the core.

The amount of radioactivity released from the core materials during any accident would depend upon the length of time and the extent to which the core was overheated. For accidents in which the cooling was soon restored to an adequate level, radioactive releases could be expected to be relatively small. However, for accidents in which the system was either slowly or never restored to adequate cooling conditions, the radioactive releases could be substantial.

This report considers the entire spectrum of accidents from those involving relatively little core damage, and therefore small releases of radionuclides, to those involving potentially large fractions of the radioactive content of the core, and therefore large radionuclide releases. This chapter reviews those factors which affect the magnitudes and the rates of the radioactive releases resulting from such accidents. (More detailed descriptions of the qualitative factors affecting the source terms for such accidents are presented in Appendices B-D of the other report for this project [Niemczyk and McDowell-Boyer, 1982].)

2.2 RELEASES FROM CORE MATERIALS

During any accident involving core damage, radioactivity would be released from the core materials into the reactor pressure vessel (RPV). In addition, in accidents in which the core melted through the RPV, radioactivity could be released directly into the containment.

2.2.1 Releases Inside the Reactor Vessel

Releases of radionuclides from the core materials could occur as the result of several different processes. The relative importance of each of these processes in determining the overall releases would depend on the nature of the processes, the course of the accident, and the radionuclides being considered.

Most accidents of concern in this report would result in a loss of coolant water from the reactor so that at least part of the core would become uncovered. This, in turn, would permit overheating of the core. If, as would be typical, the region around the core were filled with water vapor as the core were heating up, then steam-metal reactions would occur, resulting in the evolution of substantial quantities of both hydrogen and heat. The heat released by such reactions would be on the order of the decay heat of the reactor and thus could substantially increase the rate of heating of the core. The reducing atmosphere provided by the hydrogen* and the steam would affect significantly the chemical forms of some of the released radionuclides.

In contrast, if the region around the core were dry during heatup, then only the decay heat from the core would be available for heating and so the rate of heatup would be relatively slow. In addition, if the region were dry, the atmosphere in the core would not necessarily be a reducing one. Thus different chemical forms might be expected.

As the core uncovered and heated up, the fuel rod claddings would rupture, allowing the release of the gases which had accumulated in the fuel-cladding gaps. Such releases would consist primarily of noble gases and other volatile species. For a given rod, this release would be essentially instantaneous. However, inasmuch as different rods could be expected to burst at different times, the overall fuel-cladding gap release would occur over a period of time. That release would be expected to form a relatively small component of the total radioactive release for any meltdown accident but could form a significant part for other lesser accidents.

The rupture of the cladding also would permit radionuclides which diffused or otherwise escaped from inside the fuel rods after rupture to be released. As the temperatures increased, the rate of this release, and hence the amount of material released in this manner, would also increase. This "diffusion" release would be relatively small for most meltdown accidents but could be significant in accidents involving substantial periods of elevated core temperatures somewhat less than those resulting in melting.

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*The heated zirconium in the fuel rod cladding would be the initial source of reduction.

As the temperatures increased farther, the fuel would begin to melt. During the melting, some of the more volatile components could be evaporated from the various liquid surfaces. Both structural and core materials would be included in this release. The slower the overall process was, the more structural materials would be melted and incorporated into the corium (molten core plus structural materials).

As the accident progressed further, some or all of the melted materials could fall ("slump") or otherwise move into the bottom of the RPV. If the molten material formed a single mass there, the rates of releases of all the radionuclides might be substantially decreased because of the much smaller relative surface area of the material in the bottom of the reactor vessel than in the fuel rods.

Because of the large temperature gradients within the core during most accidents, the core would melt on a region by region basis. Thus, like the gap release, the total release of any given radionuclide or other material would occur over a period of time. However, in general, the more volatile radionuclides would tend to be released early in the heatup and melting. In fact, some of them would be totally released long before the core completely melted.* Consequently, large quantities of such radionuclides could be released even for accidents involving less damage than complete meltdown. In contrast, the less volatile radionuclides would tend to be released toward the end of melting. Therefore, they would not be released in large quantities unless substantial melting of the core occurred. In a complete meltdown, their releases would continue after RPV meltthrough. Indeed, the largest quantities of at least some of them could be released after that meltthrough.

Any time after the fuel cladding failed, water might contact the core materials. For any accident terminated prior to complete melting of the core, this would always be the situation. Because of the elevated temperatures of the fuel at the time of contact, significant fragmentation of a sizeable fraction of the fuel might occur. Both the initially elevated temperatures and the large surface areas resulting from fragmentation could cause substantial leaching of some of the radionuclides to occur very rapidly. Such a leach release would be highly sequence and element dependent. It could be especially important for accidents involving both substantial cladding failure and less than complete melting of the core.

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*In the case of a complete meltdown, it is conceivable that if melting in the core region occurred rapidly enough, a certain fraction of the core might not even melt. Thus, that fraction would not be available for some types of radionuclide releases.

2.2.2 Releases Outside the Reactor Vessel

If the melting continued, the corium would melt its way through the bottom of the reactor vessel and fall onto the containment basemat. The interaction of the melt with the concrete basemat could result in the release of large quantities of materials into the containment atmosphere. Both sparging of gases through the melt and evaporation of materials from the melt would be expected to be significant for such release of materials, with sparging generally being more important. Not only structural and core materials but also concrete decomposition products would be included in this release, with the latter component dominating the mass of the release.

If water were present in the reactor cavity at the time of RPV melt-through, the melt might be quenched and so the releases of radionuclides might be delayed until the water had boiled or drained off the melt. At such time, however, the releases could resume. Eventually, the molten mass would cool due to both loss of decay heat and incorporation of structural materials and concrete residua. Although complete cooling could take months or years, it would probably only take several hours at most for the mass to cool sufficiently that further radionuclide evolution would be negligible. Therefore, evolution of radionuclides would taper off relatively rapidly after RPV melt-through.

Another possibility if water were present in the reactor cavity at the time of RPV melt-through would be a steam explosion. Such an explosion could result in the rapid production and release to the containment of finely divided corium particles from a large fraction of the mass of the corium. The release of many radionuclides would be significantly enhanced by such particle formation due to the increased surface area. Alternatively, such a steam explosion could result in the scattering of substantial masses of the corium from the main corium mass. The scattered masses might be cooled more rapidly than the main mass and consequently the potential for release of some radionuclides could be decreased.

Relative to the reducing atmosphere typically found in the core region, the atmosphere in the containment might have more of an oxidizing character.* Thus the chemical forms of certain species released in the containment might be much different than those released in the core region. In addition, an oxidizing atmosphere would greatly enhance the release of certain radionuclides if a steam explosion occurred.*

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*This would not apply to the atmosphere found inside inerted drywells of Mark I BWR's (boiling water reactors) before containment failure.

2.3 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

During an accident, radionuclides released from the core region would enter the primary coolant system (PCS). In many accidents, such radionuclides would be swept by steam and/or water flows from the PCS into the containment. However, in some accidents, significant amounts of those radionuclides could be retained in the PCS as the result of various types of interactions. For example, any released gases could absorb or condense onto particulates and coolant system surfaces, react chemically with other species in the coolant system atmosphere or with coolant system surfaces, or dissolve in and/or otherwise react with any water present in the system. The particles released from the core could agglomerate onto other particles and even eventually be removed from suspension, plateout onto coolant system surfaces by various processes, or react chemically. Any of the removed material could subsequently be resuspended, revaporized, and/or otherwise re-released so that it could be entrained in the coolant system fluids and subsequently transported out of the PCS. (See Table 2.1 for a detailed review of the major natural processes in the PCS and their effects.)

A major means of retention in the PCS for certain radionuclides in some accidents would involve aerosol phenomena. In particular, if the accident conditions were such that high concentrations of aerosols were present, those radionuclides residing on or in particles could be retained in the PCS as the result of various aerosol processes. Inasmuch as removal by such processes could increase disproportionately with the total aerosol concentration, those accidents in which large quantities of aerosols would be generated could frequently be affected much more noticeably than those in which only small quantities were produced.

Whether a given radionuclide was present as an aerosol would depend on its volatility. Thus, in part, the retention in the PCS would depend on the volatility of the radionuclide. In general, the more volatile elements would tend to enter the PCS and transport through it as gases while the less volatile elements would tend to rapidly condense into aerosol particles and transport as such. The physical phases of the elements of intermediate volatility would be functions of the thermal conditions in the PCS.

The extent of retention of any radionuclide in the PCS would depend not only upon the volatility of the radionuclide but also upon several

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*A steam explosion also could occur during the slumping of the molten core into the bottom of the RPV. Inasmuch as the atmosphere in the RPV would typically not be an oxidizing one, the enhancement of the release of certain radionuclides would probably not be as large as in a comparable steam explosion in the containment.

Table 2.1. Effects of natural processes in primary coolant system during accidents involving core damage

Process	Species	General effects	Effects on radionuclides escaping from primary coolant system	
			Limited-core-damage accidents	Meltdown accidents
Agglomeration (Brownian; gravitational; turbulent)	Aerosols	Increases particle weights	The low aerosol concentrations typical of some accidents would not be rapidly decreased by agglomeration and deposition; however, the high aerosol concentrations in some accidents could result in substantial agglomeration and deposition if steam velocities through the PCS were low.	High aerosol concentrations could be rapidly decreased by agglomeration and deposition for those accidents in which low steam velocities through the PCS occurred.
Steam condensation	Aerosols	Increase particle weights; compacts individual particles	Due to the relatively low temperatures in the PCS, the aerosols might be affected by condensing steam.	Due to the relatively high temperatures in the PCS, the aerosols generally would not be affected by condensing steam.
"Condensation" onto particles (evaporation from particles) (condensation; adsorption;...)	Gases; aerosols	Increases particle weights; decreases gas concentrations; changes aerosol composition	Due to the relatively low temperatures in the PCS, vapor deposition could be substantial for all compounds.	Due to the relatively high temperatures in at least parts of the PCS, vapor deposition would affect only compounds of intermediate and low volatility in much of the PCS.
Deposition (gravitational; diffusional; thermophoretic; diffusiophoretic)	Gases; aerosols	Decreases gas or aerosol concentration	The low aerosol concentrations typical of some accidents would not be rapidly decreased by agglomeration and deposition;[a] however, the high aerosol concentrations in some accidents could result in substantial agglomeration and deposition if steam velocities through the PCS were low.	High aerosol concentrations could be rapidly decreased by agglomeration and deposition for those accidents in which low steam velocities through the PCS occurred.

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Table 2.1 (continued)

Process	Species	General effects	Effects on radionuclides escaping from primary coolant system	
			Limited-core-damage accidents	Meltdown accidents
Resuspension	Aerosols	Increases aerosol concentration	Probably not important.	Might be large for accidents with substantially varying steam flow rates through the PCS.
"Condensation" onto surfaces (evaporation from surfaces) (condensation; adsorption;...)	Gases	Decreases gas concentration	Due to the relatively low temperatures of surfaces in the PCS, vapor deposition could affect compounds of all volatilities both inside and outside the RPV.	Due to the relatively high temperatures of the surfaces of the PCS, vapor deposition generally would affect only compounds of intermediate volatility and would occur only outside the core region.[b]
Electrostatic interactions	Aerosols	May hinder deposition	Probably not important.	Probably not important.
Radioactive decay	Gases; aerosols	Changes chemical composition of gases and aerosols; heats aerosols	Chemical changes potentially important.	Chemical changes potentially important.
Chemical reactions	Gases; aerosols	Changes chemical compositions of gases and aerosols; may decrease gas concentration	Potentially important (for example, formation of CsI and reaction of Te with PCS surfaces).	Potentially important (for example, formation of CsI and reaction of Te with PCS surfaces).
Scrubbing (diffusiophoresis; ...)	Gases; aerosols	Decreases concentrations	Passage through standing water in the PCS could substantially reduce both gas and aerosol concentrations exiting from the PCS.	Passage through standing water in the PCS could substantially reduce both gas and aerosol concentrations exiting from the PCS.

[a] The relatively large particles ejected during the gap release might be deposited in the PCS.

[b] This assumes that compounds of low volatility would condense into aerosols before reaching any surfaces suitable for condensation.

accident characteristics, including the length of the path through the system, the temperatures of the fluids and surfaces in the system, the velocities of gases and particulates through the system, the rate of release of aerosols into the system, and the presence (or absence) of water in the dominant pathway(s) through the system. Shorter physical paths, higher temperatures, higher velocities, and lower aerosol generation rates would usually tend to decrease the likelihood of retention in the coolant system for most species. In contrast, the presence of water would increase the amount of retention in the coolant system of most species.

The dominant paths which the radionuclides would follow through the coolant system generally would be determined by the location of the pipe break in the case of a loss-of-coolant accident (LOCA) or by the nearest relief or safety valve in the case of a transient-initiated event. Large hot-leg pipe breaks usually would result in the shortest paths to the containment (or drywell) and thus the largest releases of radioactivity escaping to the containment, if other conditions were equivalent.

The temperatures in the primary coolant system would depend upon the rate of heatup and/or melting, with sequences having more rapid heatup (melting) frequently being expected to have both higher temperatures and larger temperature gradients in the coolant system. (If heatup [melting] occurred more slowly, then some of the heat added to the coolant system would be dissipated to other locations during the course of the accident.)

High steam temperatures in the PCS would tend to prevent the condensation of compounds of high and intermediate volatility onto suspended aerosols; likewise, high surface temperatures in the PCS would tend to discourage the condensation of those compounds onto surfaces. Thus high temperatures in the PCS would tend to preclude condensation of large quantities of both highly and moderately volatile compounds there. For accidents involving less than complete meltdown of the core, the temperatures in the PCS, especially in those regions far from the core region, would be relatively low and so might permit deposition of large amounts of such compounds. For accidents involving meltdown, the temperatures in at least some of the PCS would generally be high and would tend to discourage condensation of such compounds in those regions. (Although condensation in the PCS might occur early in such an accident, the deposited radionuclides would often be re-evaporated and the flushed from the PCS as it heated up during the accident. Alternatively, however, such radionuclides might react with the PCS surfaces as they heated up and thereby be retained.)

In regions of relatively low temperatures, whether deposition occurred onto aerosols or onto surfaces would always depend in part on the relative surface areas presented by those two media. If deposition were onto aerosols, the radionuclides could still be swept from the PCS with the aerosols. However, if deposition were onto surfaces, permanent retention in the PCS would often be more likely.

The velocities of radionuclides through the coolant system would depend upon the driving forces. If water were present around the core during overheating, the large quantities of hydrogen and steam evolved during heatup would tend to rapidly flush any radionuclides from the core region through the coolant system. In contrast, if water were not present, only expansion of the gases due to heating would drive the radionuclides from the core region; this process would be relatively slow, with substantial fractions of the generated radionuclides often not escaping into the containment until long after their initial releases. In many accidents involving meltdown, steam flows and velocities through the coolant system would be relatively high both early in the accident as the water was being boiled off from around the core and later in the accident as water was being evaporated out of the bottom of the RPV due to slumping of the melt into it. The steam flow rates and velocities at other times would be highly dependent on both the accident scenario and the reactor design.

LOCA's, especially ones involving large pipe breaks, often would have relatively large steam velocities and therefore short radionuclide residence times in the PCS. The retention of aerosols in the PCS would frequently be small for such accidents. In contrast, some transient-initiated accidents and certain small LOCA's typically would have low steam velocities, and consequently long radionuclide residence times, in the PCS. For some such accidents, the delay could be such that the majority of the radioactivity would not even be released to the containment unless and until rupture of the coolant system occurred, for example, by RPV melt-through. The retention of aerosols and some gases in the PCS could be significant in such accidents.

The aerosol generation rates would depend upon the rate of release of materials into the PCS. Those materials potentially forming aerosols would be composed of moderately and less volatile radionuclides, control rod components and structural materials. Higher aerosol generation rates would tend to result in higher aerosol concentrations in the PCS and thus could promote sharply enhanced removal of aerosols within the PCS.

The presence of water in the dominant flow path(s), for example, in the pressurizer quench tank in a transient-initiated accident in a pressurized water reactor, would permit scrubbing of some of both the more soluble gases and the aerosol particles from the fission product stream. In addition, it might provide an obstacle to the flow to the containment for all radionuclides.

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2.4 TRANSPORT THROUGH THE CONTAINMENT

The materials entering the containment indirectly via the primary coolant system would consist of mostly core structural materials and fission products. In contrast, the materials entering the containment directly would be composed primarily of concrete residue, along with much smaller amounts of fission products and structural materials. Natural processes, as well as the operation of ESF's, would affect the removal of radionuclides from the containment atmosphere.

2.4.1 Natural Processes

Both the gases and the particulates released to the containment would undergo basically the same natural processes as the materials initially released to the coolant system. (See Table 2.2 for a review of the major natural processes in the containment and their effects.) The principal differences would be due to the dissimilarities in the physical and chemical conditions present in those two places. The conditions in the containment would depend upon both the accident initiators and the performance of various ESF's. These, in turn, would be very accident specific.

If the core melted through the RPV, the region just above the melt would be very hot. However, in general, for any accident the temperatures in most of the containment would be much lower than those in either the core region or the primary coolant system. Thus more species would be present as aerosols in the containment than were present as such in the coolant system. However, due to both retention in the coolant system and dilution in the containment atmosphere, the total aerosol concentrations in the containment generally would be much lower than those in the coolant system. (They might not be lower if aerosol residence times, and therefore the aerosol concentrations, in the PCS were relatively small.) As a result, the concentration-dependent removal processes which frequently dominate the behavior of the aerosols, and thus the behavior of the associated radionuclides, in the coolant system often would be much slower in the containment.

2.4.2 Effects of Engineered Safety Features

During any accident, various ESF's might operate (see Tables 2.3 and 2.4 for a review of the major ESF's and their effects). For any accident, one of the most important ESF's would be containment leak-tightness. The containment would tend to confine the radioactivity and thus permit time for both natural processes and processes due to the operation of the other ESF's to remove radioactivity from the containment atmosphere. Consequently, an unbreached containment usually would result in substantially lower amounts of most radionuclides being released to the outside environment. For example, if severe rupturing of the containment occurred early in the accident scenario, a large fraction of all the radionuclides released to the containment could escape to the environment

Table 2.2. Effects of natural processes in containment during accidents involving core damage

Process	Species	General effects	Effects on radionuclide concentrations in containment atmosphere	
			Limited-core-damage accidents	Meltdown accidents
Agglomeration (Brownian; gravitational; turbulent)	Aerosols	Increases particle weights	The low aerosol concentrations would not be rapidly decreased by agglomeration and deposition.	The high aerosol concentrations generated during some accidents could be rapidly decreased by agglomeration and subsequent deposition.
Steam condensation	Aerosols	Increases particle weights; compacts individual particles.	The low aerosol concentrations would be more rapidly decreased by condensing steam.	The high aerosol concentration generated during some accidents would be more rapidly decreased in high steam concentrations, especially if accompanied by cooling.
"Condensation" onto particles (evaporation from particles) (condensation; adsorption...)	Gases; aerosols	Increases particle weights; decreases gas concentrations; changes aerosol composition.	Compounds not previously deposited in the PCS, especially ones of high volatility, could be deposited.	Compounds not previously deposited in the PCS, especially ones of high or intermediate volatility, could be deposited.[a]
Deposition (gravitational; diffusional; thermophoretic; diffusiophoretic)	Gases; aerosols	Decreases gas or aerosol concentration	The low aerosol concentrations would not be rapidly decreased by agglomeration and deposition.	The high aerosol concentrations generated during some accidents could be rapidly decreased by agglomeration and subsequent deposition.
Resuspension	Aerosols	Increases aerosol concentration	Probably not important.	Probably not important.
"Condensation" onto surfaces (condensation; adsorption...)	Gases	Decreases gas concentration	Compounds not previously deposited in the PCS, especially ones of high volatility, could be deposited.	Compounds not previously deposited in the PCS, especially ones of high or intermediate volatility, could be deposited.[a]

Table 2.2 (continued)

Process	Species	General effects	Effects on radionuclide concentrations in containment atmosphere	
			Limited-core-damage accidents	Meltdown accidents
Electrostatic interactions	Aerosols	May slow down rate of deposition or increase rate of resuspension	Probably not important.	Probably not important.
Radioactive decay	Gases; aerosols	Changes chemical composition of gases and aerosols; heat aerosols	Chemical changes probably not important; aerosol heating may affect steam condensation.[b]	Chemical changes probably not important; aerosol heating may affect steam condensation and aerosol deposition.[b]
Chemical reactions	Gases; aerosols	Changes chemical composition of gases and aerosols; may decrease concentrations	Potentially important.[c]	Potentially important.[c]
Evaporation	Dissolved species	Increases gas concentration	Evaporation from water (and surfaces) after the accident could form a long-term source of some radionuclides.	Evaporation from water (and surfaces) after the accident could form a long-term source of some radionuclides.

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[a] Vapors of low volatility fission products released directly into the containment after RPV meltthrough would typically condense into aerosols shortly after being vaporized from the melt.

[b] A notable example is the set of reactions which result in the formation of organic iodides. Another example is the set of reactions which might occur due to hydrogen burning.

[c] Powers (1982).

Table 2.3. Effects of selected engineered safety features during accidents involving core damage[a]

Engineered safety feature	Plant	Electricity required	Effects	Effect on radionuclides in containment during limited-core-damage accidents	Effect on radionuclides in containment during complete meltdown accidents
Containment leak-tightness	all	no	Prevents radioactivity in containment from reaching environment; leakage often permits some release	Retards release to environment, permitting removal and decay mechanisms to lessen the releases	Retards release to environment, permitting various removal and decay mechanisms to lower the radionuclide concentrations in the containment; severe early containment rupture substantially lessens benefits of containment
Containment spray system	PWR's (restricted to upper compartment in ice condenser PWR's)	yes	Reduces pressure in containment; removes radioactivity and heat from containment atmosphere; promotes radionuclide removal by both washout and acceleration of aerosol removal processes	Removes some aerosols[b] and some iodine[c,d]	Large aerosol concentrations would be acted on most effectively;[b] very high aerosol concentrations could clog spray system
Containment recirculation filter system	some PWR's	yes	Removes radioactivity from containment atmosphere and cools containment atmosphere	Traps aerosols and some iodine[e]	Large aerosol concentrations in the containment could clog the filters and rapidly render them ineffective
Auxiliary building filter system	PWR's	yes	Removes radioactivity from auxiliary building atmosphere	Traps aerosols and some iodine[e]	Large aerosol concentrations in the auxiliary building could clog the filters and rapidly render them ineffective
Main stream isolation valve leakage control system	BWR's	yes	Filters gases leaked past inboard main steam isolation valve	Traps aerosols and some iodine[e]	For severe accidents, this system would generally be by-passed; when not by-passed, large aerosol concentrations could clog the system

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Table 2.3 (continued)

Engineered safety feature	Plant	Electricity required	Effects	Effect on radionuclides in containment during limited-core-damage accidents	Effect on radionuclides in containment during complete meltdown accidents
Pressure suppression pool	BWR's	no	Reduces pressure in containment (condenses steam released from drywell); scrubs some radioactivity from the drywell releases	Removes some aerosols[b] and some iodine[c]	Large aerosol concentra- would be acted on most effectively;[b] probably most effective if pool temperature is below boiling
Standby gas treatment system	BWR's	yes	Filters contaminants leaked from primary containment into secondary containment	Traps aerosols and some iodine[e]	For severe accidents, this system would generally be by-passed; when not by-passed, large aerosol concentrations could clog the system
Ice bed condenser	Ice-condenser PWR's	no (yes for long-term)	Reduces pressure (condenses steam) in containment; scrubs some radioactivity from the atmosphere; removes molecular iodine	Removes some aerosols[b] and some iodines[c,f]	Large aerosol concentrations would be acted on most effectively;[b] effective only if ice beds had not melted

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[a] For more extensive discussions of the effectiveness of various engineered safety features, see both Chapter 8 of NUREG-0772 (USNRC, 1981) and Chapter 4 of NUREG-0771 (Pasedag, Blond, and Jankowski, 1981).

[b] The removal rate is greater for large particles than for small particles. Inasmuch particles in high density aerosols tend to agglomerate into larger particles, removal rates by sprays usually can be expected to be relatively greater for situations with higher aerosol concentrations.

[c] Organic iodine is not removed very effectively by this ESF.

[d] At some reactors, chemical additives help to convert elemental iodine to less volatile iodides.

[e] Organic iodine is not trapped as effectively as either elemental or particulate iodine.

[f] Chemical additives help to convert elemental iodine to less volatile iodides.

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Table 2.4. Presence of selected engineered safety features

Engineered safety feature	PWR	Ice condenser PWR	BWR-Mark I	BWR-Mark III
Containment leak-tightness	Yes (large volume, high pressure)	Yes (medium-large volume, medium pressure)	Yes (small volume, high pressure)[a]	Yes (medium-large volume, medium pressure)[b]
Containment spray system	Yes	Upper compartment only	No	Yes, but not necessarily activated
Containment recirculation filter system	Some	Yes	No	No
Auxiliary building filter system	Yes	Yes	No	No
Main steam isolation valve leakage control system	No	No	Yes	Yes
Pressure suppression pool	No	No	Yes	Yes
Standby gas treatment system	No	No	Yes	Yes
Ice-bed condenser	No	Yes	No	No

[a] Mark I's (and Mark II's) have small-volume drywells and large-volume secondary containments; whereas the drywells are designed to withstand high pressures (50-60 psig), the secondary containments are not designed to endure elevated pressures (design pressures are approximately 0.25 psig). (In a Mark I or a Mark III BWR, the primary containment is composed of the drywell and the wetwell; the secondary containment is a shield building.)

[b] Mark III's have small-volume drywells and large-volume primary and secondary containments; both the drywells and the primary containments are designed to withstand moderate pressures (15-25 psig). The secondary containment (or shield building) is not designed to endure elevated pressures.

than a breached one. In contrast, if severe rupturing occurred much later in the scenario, then large fractions of only the more volatile and less reactive radionuclides, for example, the noble gases, might escape. Likewise, if only a relatively small leakage from the containment occurred throughout the accident, then significant fractions of only the less reactive, more volatile species would tend to escape.

Among the other ESF's which might affect the amounts of radioactivity escaping would be containment sprays, suppression pools, and ice-bed condensers. During accidents, operation of any of these features would tend both to lower the pressure in the containment and to remove radioactivity from the containment atmosphere. Consequently, any of these features would tend to lower the amounts of radioactivity released to the environment outside the nuclear plant. These various types of pressure suppression systems would, in general, be effective in substantially lowering those amounts escaping to the outside provided these systems were still operating during the releases from the core materials and/or from the PCS. The major exceptions would be those accidents involving early catastrophic breaching of the containment and those accidents in which the dominant radionuclide pathways would bypass the containment. For consideration of the environment inside the plant, the primary effect of such ESF's would be to redistribute the radionuclides within the containment. The various suppression systems would tend to transfer the radionuclides from the containment atmosphere to the water in the reactor building and to a lesser extent to the surfaces in the system.

Other ESF's, such as the containment recirculation filter systems and the standby gas treatment (filter) systems, would tend primarily just to remove radioactivity from either the containment atmosphere or from the escaping gases. These ESF's would be most effective for accidents in which the containment remained unruptured. In contrast to the pressure suppression ESF's, the filter systems would usually be effective in substantially lowering the total amount of the radioactivity released only for less severe accidents. In general, the filter systems would tend to localize some of the activity in the filter systems themselves.

2.5 DIFFERENCES AMONG REACTORS

The magnitudes and the probabilities of the various releases which might occur at a given site are both dependent on certain features of the nuclear plant under scrutiny. In particular, they depend on factors including the size and the status of the core, as well as the engineered safety features (ESF's) of that plant (see Table 2.4). In addition, they depend upon the basic designs of the core, the coolant system and the containment(s). Furthermore, they are a function of the detailed construction characteristics of the plant. All of these factors vary widely among plant.

As a result of all these differences, the magnitudes and the probabilities of the potential accidental releases differ substantially from plant to plant. Indeed, the types of accident sequences which dominate the risk to any population-of-concern at certain plants are frequently negligible contributors to the risk to the analogous populations at other plants. Furthermore, the anticipated relative magnitudes of the radionuclide releases resulting from a given type of accident sequence vary greatly among plants. Likewise, the probability for a given type of accident sequence varies widely even among apparently similar plants.

2.6 ACCIDENT SPECTRUM

Of the large number of accidents which might occur at any plant, only a very limited number of types of these accidents are typically important for considering either the worst possible consequences or else the overall risk to any population-of-concern. For example, in a design basis accident approach such as that used for equipment qualification, the only important accidents in the accident spectrum are those which result in releases of radionuclides which bound the possible releases for all accidents with the design basis. In a risk assessment approach such as that used for certain emergency planning considerations, the only important accidents are those which dominate the risk, that is, those whose combined likelihoods of occurrence and potential consequences make them among the riskiest.* In contrast, a much larger number of types of accidents are typically relevant for considering the potential consequences of all possible individual accidents to the same population-of-concern. Thus, for detailed emergency planning considerations for individual accidents, a broad, relatively detailed spectrum of accidents must be investigated.

The ranges of each of the accident spectra are delimited by their worst-case and their best-case accidents. For equipment qualification, the worst case for most equipment involves partial meltdown of the core, rapid airborne transport through the coolant system, and limited leakage from the containment building.** The best case involves minor core damage, slow waterborne transport through the coolant system and extensive leakage from the containment. In contrast, for emergency planning, the worst case involves a complete meltdown of the core accompanied by substantial breaching or circumventing of the containment early in the accident. The best case involves minor core damage accompanied by very limited leakage from the containment.

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*This assumes that the bounding cases and the risk-dominant sequences can be easily identified. If this is not the situation, then a larger portion of the spectrum of possible accidents must be investigated.

**For at least some monitoring equipment, the worst case would involve complete melting of the core, rapid transport through the coolant system and limited leakage from the containment. However, such equipment is not considered in this report.

The relative importance of any accident in a design basis spectrum such as that considered for equipment qualification depends on only the magnitude of the resultant radionuclide releases. The relative importance of any accident in a risk assessment spectrum such as that considered for certain emergency planning purposes depends on both its probability of occurrence and the magnitude of the resultant radionuclide releases. In contrast, the relative importance of any accident in a spectrum such as that considered for emergency planning for individual accidents depends only on the magnitude of the resultant radionuclide releases.

2.7 SUMMARY

The fraction of any radionuclide released from the core materials during an accident could range from a negligible one to nearly the entire core inventory, depending on the radionuclide and the accident scenario. More volatile radionuclides, such as noble gases and halogens, would always tend to be released more completely than less volatile ones. In meltdown accidents, almost the entire inventory of some of the more volatile radionuclides would be released before RPV melt-through while only small fractions of the less volatile radionuclides would be released by that time. Larger amounts of some of the less volatile materials could be released after that time, although the total fractions of those materials released typically would be small. Accidents involving less heating of the core materials would tend to result in smaller releases of all radionuclides.

In general, radionuclides present as gases would tend to pass through the primary coolant system unaffected, except for radioactive decay, although some of those radionuclides could undergo substantial retention in the PCS as the result of condensation or chemical reactions. In contrast, most of the radionuclides present as aerosols could undergo substantial attenuation if there were large delays in the movement through the PCS, with the fraction of any aerosol retained in the PCS ranging from a very small one to a substantial fraction, depending on the accident. In particular, the amount of retention would depend on both the concentration of the aerosol and the length of time the material was contained in the PCS. In general, large LOCA's would tend to result in negligible aerosol retention due to delays while some transients and some small LOCA's would result in significant delays and hence potentially substantial retention in the PCS. Retention in the PCS of both gases and aerosols could also be large in those accidents in which the radionuclides had to pass through standing water to reach the containment.

The fractions of the radionuclides ultimately escaping from the containment would depend on the status of the containment when the radionuclides reached it, as well as on the effects of the ESF's such as containment sprays. Typically, accidents either with bypassing of the containment or with early catastrophic containment failure would result in the largest releases to the environment. Of those accidents with early containment failure, the ones with early failure of the radionuclide removal systems would result in the largest releases. In such accidents, natural removal processes would have little time to affect the amounts of radioactivity escaping.

The accident spectrum for equipment qualification for most equipment consists of accident sequences involving core damage less than complete melting. For a design basis approach, only a very limited number of those accident sequences are considered. The sequences important to the equipment vary from plant to plant.

The accident spectrum for emergency planning consists of both meltdown and limited-core-damage accident sequences. For either a risk assessment or a planning evaluation for all possible accidents, a large number of types of accident sequences must be considered. The specific sequences dominating the risk to the human population and those important to any planning evaluation vary substantially from plant to plant.

References for Chapter 2

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3. ASSUMPTIONS AND PROCEDURES

3.1 INTRODUCTION

The discussions in the previous chapter focused on reviewing the basic factors involved in considering the release of radionuclides from the core materials and the subsequent transport of those radionuclides through the nuclear plant. This chapter outlines the approach adopted for quantifying the aspects of those factors relevant to estimating source terms. In particular, this chapter describes most of the basic "assumptions" and procedures determined to be appropriate for estimating source terms. Chapter 3 of the other report for this project (Niemczyk and McDowell-Boyer, 1982) summarizes the corresponding methods used to obtain numerical estimates for the illustrative calculations for this project.

The appropriate overall approach for quantifying source terms is outlined in Section 3.2 of this chapter. Next, specific assumptions and procedures for estimating the source terms for any individual accident are described in Section 3.3. The procedures for describing the quantities of radionuclides released from the core materials and the rates of those releases are presented in the first subsection of Section 3.3. Then, the assumptions for estimating the retention of radioactivity in the primary coolant system are discussed in the next subsection. Finally, the procedures for quantifying the releases of radioactivity from the containment to the environment are reviewed in the last subsection of Section 3.3.

The appropriate approaches for grouping accident sequences into accident classes and for estimating the resultant source terms for those classes are described next. In particular, the assumptions for defining the accident spectra appropriate for equipment qualification and for emergency planning are presented in Section 3.4.

Some of the basic sources of uncertainty in the assumptions are indicated in Section 3.5. Finally, the adopted assumptions are summarized in Section 3.6. All of the assumptions and the factors affecting their utilization are considered in much greater detail in Appendices B through D of the other report for this project (Niemczyk and McDowell-Boyer, 1982).

3.2 OVERALL APPROACH

The emphasis of this work was to determine and demonstrate the use of the "best" available methods for estimating radionuclide releases and behavior for LWR accidents.* In general, the state-of-the-art of certain aspects of the estimation of source term magnitudes involves the use of complex computer codes and, in some cases, the subsequent development of estimates from the outputs of those codes. Due to the limited scale of this project, it was not possible to describe in detail all the assumptions implicit in such codes. In addition, it was not possible to mention all the necessary caveats associated with the use of such codes. As a result, the overall approach taken here to indicate the appropriate assumptions and procedures is to briefly describe model(s) and/or computer code(s) which implement them. In those cases in which the assumptions cannot be simply described and/or are not exemplified in any models or codes, the models or codes or procedures most appropriate to use are indicated. Adequate usages of these models and codes are demonstrated by illustrative calculations. In addition, many problems associated with their usage are discussed in detail in the appendices of the other report for this project. Where it is thought advisable to modify these models or the outputs of these codes either to reflect acknowledged shortcomings or to reflect recently gained insights into the release and transport of radionuclides during and after nuclear reactor accidents, the details are so indicated in the appendices of the other report.

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*Although source term assumptions can be interpreted to be procedures and methods as they are in this chapter, they might also be

3.3 INDIVIDUAL ACCIDENTS

To describe quantitatively the source terms for any specific accident, the primary quantities of interest are the amounts of the initial releases of the radionuclides and other materials, the fractions of all those materials escaping from the primary coolant system, and the fractions of the radionuclides ultimately escaping from the containment. Both sets of escape fractions depend not only on the amounts, but also on the rates, of the initial releases. Assumptions and procedures appropriate for estimating all these quantities are given in this section.

For all source term considerations, the radionuclide classification scheme of the Reactor Safety Study (RSS; USNRC, 1975) is adopted, unless element-dependent data are available. In the RSS scheme, the radionuclides are grouped into seven basic element categories (see Table 3.1). Each radionuclide is assumed to be released and transported according to the basic properties (mainly volatility) of its element category. Except where mentioned otherwise, the detailed chemistries of the various radionuclides are not explicitly considered.

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interpreted to be estimated values for the anticipated radionuclide releases. Because of the many problems associated with the use of intricate models and computer codes, this latter interpretation may be more appropriate for certain regulatory purposes. Thus, estimates such as those represented by the illustrative calculations presented in the following two chapters may be more suitable for many general uses than the procedures outlined here. See the related discussions in Chapters 4 and 5.

3.3.1 Releases from Core Materials*

The procedures adopted for estimating releases of radionuclides from the core materials before reactor vessel failure are basically the same as those presented in the Technical Bases Report (USNRC, 1981). In contrast, the procedures for estimating releases of radionuclides to the containment after reactor vessel failure are based on the methods presented in the Reactor Safety Study. The procedures for estimating releases of aerosols both before and after vessel failure are essentially those given in the Technical Bases Report.

In the Technical Bases Report, before vessel failure, the sum of all the release processes is assumed to result in a net release rate for any radionuclide which is dependent only on the temperature of the material containing the radionuclide (Wichner, Kress, and Lorenz). The indicated element-dependent release rates are obtained from fits of empirical information.** The best current values for these temperature-dependent rates are essentially those presented in the Technical Bases Report, with the exception of a few modifications stated in Appendix B of the other report for this project. The rates in this procedure are assumed to include the releases by all considered processes except leaching and "vaporization" (i.e., sparging). [Thus, the procedure adopted here differs from that in the RSS in that the contributions of the various processes (e.g., gap release, meltdown release, etc.) to the total releases are not treated as discrete events.]

In the overall procedure used here, estimates of the releases of the various radionuclides before vessel melt-through as a function of time for a given accident sequence are obtained by combining the temperature-dependent release rates with temperature histories of the different core regions. The required core material temperatures in various regions of the core, as functions of time, are taken from thermal-hydraulic estimates obtained from a code such as MARCH (Wooton and Avci, 1980) for meltdown accidents and a code such as RELAP (Aerojet Nuclear Company, 1976) for accidents without melting of the core.

According to the Technical Bases Report, to estimate the total amounts of aerosols released before vessel failure, structural materials and cladding are treated in basically the same manner as the radionuclides (Wichner, Kress, and Lorenz, 1981). Aerosol production is

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*See the discussions in Appendix B of the other report for this project for detailed consideration of the application of the procedures outlined here.

**In this approach, radionuclide chemistry need not be specified in order to obtain release fractions as a function of time during the initial stages of an accident. This approach assumes that the experimental data are representative of conditions existing during an accident in a reactor, and involves extrapolation from bench-scale to full-scale conditions.

Table 3.1. RSS element groups

Noble Gases - Xe, Kr

Halogens - I, Br

Organic Iodide

Cesium - Cs, Rb

Tellurium - Te, Se, Sb

Strontium - Sr, Ba

Ruthenium - Ru, Mo, Pd, Rh, Tc, Co

Lanthanum - La, Nd, Eu, Y, Ce, Pr, Pm, Sm, U, Np, Pu, Am, Cm, Zr, Nb estimated by assuming all radionuclides, fuel, structural materials, and cladding, except noble gases and iodine, are aerosols immediately after release. Particle sizes are ignored. Contributions to the aerosol release by molten control rod materials and structural materials beneath the core are also ignored, in lieu of pertinent data with which to quantify such releases.

An alternative method to that just described might be considered after slumping of the core into the bottom of the reactor vessel. In that other method, the rates of release of both radioactive and nonradioactive materials are taken to be reduced by a factor related to the decrease in the relative surface area of the core materials due to slumping (Parker, 1982). (See Appendix B of the other report.)

According to the procedure in the RSS, radionuclide releases after reactor vessel failure are estimated using element-dependent vaporization loss fractions. These fractions, given in the RSS, represent the losses due to gas sparging of the melt in the reactor cavity. The estimation of these releases assumes all of the gases produced in concrete decomposition pass through the melt and a core debris temperature of 3100°K during sparging. Total aerosol production after vessel failure is estimated similarly to the treatment of radionuclide aerosol production, with an empirically derived equation relating gas velocity and melt temperature to aerosol concentration in the sparging gas (Murfin and Powers, 1980; Wichner, Kress and Lorenz, 1981).*

Various accidents would differ in the temperature-time profiles of the core materials in varying degrees. Radionuclide releases should be considered for a number of representative accident sequences for the plant.

In those accidents in which leaching of the core materials also might be a relatively important release mechanism, the fractions leached should be estimated using empirical leach rates such as those of Mitchell, Goode, and Vaughn (1981). These leach fractions should be added to the fractions released directly by overheating.

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*The methodology requires an assumption of an average melt surface area in the reactor cavity during core/concrete interactions, an assumption of the fraction of gases produced that pass through the melt, and the assumption that results of the experimental design from which the empirical equation was derived may be extrapolated to full scale conditions. (The INTER subroutine of the MARCH code, which calculates core debris temperatures and gas flow rates also requires a number of assumptions that are not dealt with here.)

3.3.2 Transport in the Primary Coolant System*

No adequate general procedures have been developed for estimating the retention of radionuclides in the primary coolant system (PCS). Such retention would be expected to be very scenario and reactor dependent. In addition, estimates of such retention for many types of accidents are highly uncertain. Thus, in general, permanent retention in the PCS should not be considered for regulatory purposes for most accidents. For equipment qualification, permanent retention in the PCS should not be considered for any sequence. For emergency planning, permanent retention should be considered only for aerosol species and only for certain meltdown accidents. For both equipment qualification and emergency planning, transfer of radioactivity to water as the result of scrubbing in the pressurizer or pressurizer quench tank can also be considered for individual sequences.

The approach suggested in those cases in which potential permanent retention is considered is based on the discussions presented in Appendix C of the other report for this project. The approach in that appendix involves the use of simple assumptions about the fractions of aerosols retained based on consideration of the aerosol concentrations and the aerosol residence times in the PCS. The necessary estimates of both the concentrations and the residence times should be based on accident sequence dependent calculations of computer codes such as TRAP (Baybutt and Jordan, 1977; Jordan, Gieseke and Baybutt, 1979).

If aerosol retention is considered, the rate of aerosol generation should be estimated by the procedures given in the previous subsection, except that structural materials and cladding should not be included. In addition, the minimum mass inventory of radionuclides in the core should be used to estimate the aerosol generation rates. It should be assumed that no net aerosol removal would occur until the aerosols had reached at least the upper portion of the reactor vessel. No permanent retention larger than 0.67 should be assumed for any radionuclide for any accident sequence.

The procedure adopted to estimate removal by scrubbing in the pressurizer quench tank is basically that presented in the RSS for scrubbing by water in either the suppression pool or the reactor vessel. In that procedure, scrubbing is treated by application of a decontamination factor. For scrubbing in the quench tank, the effect is taken to apply to at most 80% of the materials released to the reactor vessel.

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*See the discussions in Appendix C of the other report for this project for detailed consideration of the application of the procedures outlined here.

3.3.3 Transport Through the Containment*

Reasonable procedures for estimating escape fractions from the containment are basically those proposed in the RSS (Postma, Owzarski and Lessor, 1975). Thus they are implemented most simply by using the computer codes which were developed during and after that study and which embody those procedures. In particular, they involve the use of CORRAL, along with its recent modifications to include containment and engineered safety features other than those considered in the RSS (Burian and Cybulskis, 1977). The thermal-hydraulic input required for CORRAL is currently best generated by appropriate use of MARCH.

Alternatively, other equivalent procedures or codes might be used. One of the important features of any appropriate set of procedures would be that it included removal of radionuclides from the containment atmosphere as the result of the operation of various engineered safety features such as containment sprays, filter systems, suppression pools, and ice-bed condensers. It could also include a limited amount of removal by natural processes for meltdown sequences. However, the rate of the removal should not exceed that observed in the Containment Systems Experiments (Hilliard, Postma, McCormack and Coleman, 1971; Hilliard and Postma, 1981).

In addition to the direct effects of the engineered safety features and the natural atmospheric removal processes, some other effects should also be considered. Among those effects is gas phase-liquid phase partitioning. Noble gases initially residing in water should be assumed to escape entirely to the containment atmosphere. Elemental iodine initially in water should be partitioned between the air and the water, according to the pH of the water. For acidic or neutral water, all the elemental iodine should be assumed to escape to the containment atmosphere. For basic water, the concentration in the containment atmosphere should be increased until it is at least 0.01 of that in the water.

Another effect which should be treated is the formation of organic iodine. Three percent of the elemental iodine initially present in the containment atmosphere should be assumed to be converted to organic iodine.

And still another effect is radioactive decay. The fractions of both noble gases and iodine escaping could be taken to be appropriately decreased by accounting for radioactive decay.

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*See the discussion in Appendix D of the other report for this project for detailed consideration of the application of the procedures outlined here.

3.4 SPECTRUM OF ACCIDENTS

3.4.1 Equipment Qualification

3.4.1.1 Method

The accident spectrum for equipment qualification includes accident sequences with only partial or delayed functioning of the emergency core cooling system. In particular, "realistic" consideration should be given to such accidents which might result in both melting of 50% of the core and the largest possible doses to the equipment. The scenarios should be such that the accidents are within, yet potentially bound, the current design basis. Accidents similar to the two described in the Technical Bases Report (Denning, 1981) are appropriate. The magnitudes of the radionuclide releases for each accident sequence are to be estimated using the assumptions and procedures described earlier, in Section 3.3 of this chapter.

3.4.2 Emergency Planning

3.4.2.1 Method

The accident spectrum for emergency planning includes accident sequences involving only minor core damage, sequences involving substantial damage to the reactor core and sequences involving complete meltdown of the core. The specific sequences considered should be such that the entire spectrum of possible releases to the environment is appropriately covered. For the most part, accidents similar to those described in the RSS but specialized to the nuclear plant under consideration would be appropriate. The magnitudes of the radioactive releases are to be estimated using the assumptions and procedures described earlier.

3.5 UNCERTAINTIES

Some of the shortcomings of the methods used here are outlined in Table 3.2. They are discussed in much greater detail for all the topics covered in this report in Appendices B-D of the other report for this project and in general in Appendix E of that report. The shortcomings result primarily from inadequate data to describe some of the relevant processes. In many cases, this lack of data has severely hampered the development of appropriate models. The potential impacts of all these problems on the source term estimates are discussed in the next chapters.

3.6 SUMMARY OF ASSUMPTIONS

The following assumptions are appropriate for estimating magnitudes of source terms for interim purposes:

Table 3.2. Problems in estimating magnitudes of source terms

Processes	Method	Problems[a]
Releases of fission products and other products from core materials and structural materials before reactor vessel failure by processes other than leaching.	Empirically-derived curves describe temperature-dependent rates for each element.	Adequate information not available for all important elements, for all temperatures, and for in-situ conditions; chemical forms largely ignored.[b]
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input not adequately known; some important processes neglected or ignored.
	("First-principles" code describes fission product releases).[c]	(Most required data not available).
Leaching of fission products.	Empirical data describes leaching.	Data not available for most elements, for an appropriate range of temperatures, and for accident (in-situ) conditions.
	("First-principles" code describes leaching).[c]	(Most required data not available).
Releases of fission products and other materials after vessel failure.	Empirically-derived release rate expression describes total aerosol generation; element-dependent vaporization fractions determine radionuclide release rates.	Limited data available for total aerosol generation rates; inadequate data available for many elements.
Transport through primary coolant system atmosphere.	"First-principles" code describes PCS transport.	Important aerosol removal processes not included; thermal hydraulic conditions not adequately known.
	"First-principles" containment codes describe aerosol processes.	Ignore thermal hydraulic conditions; ignore structure of coolant system; some data not available.
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input...

Table 3.2. (continued)

Processes	Method	Problems[a]
Scrubbing in primary coolant system.	Empirically-based decontamination factors describe scrubbing.	Adequate data not available for all important elements and conditions.
Transport through containment.	Empirically-based code describes containment transport.	Effects of potentially important removal processes may be underestimated; effects may not be correctly scaled from experimental conditions; containment rupture modes and timing not adequately predicted; thermal-hydraulic conditions not adequately known; many processes not explicitly modeled; inadequate element-dependent and condition-dependent data available.
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input.
	("First-principles" code describes containment transport).[c]	(Most required data not available; not all processes are included in any specific code.
Leakage from containment.	Empirically-based equation describes leak rate with respect to size.[c]	Mode and timing of containment failure very uncertain.

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[a] See the discussion in Appendices B-D of the other report for this project for details.

[b] To use this method, one needs either appropriate empirical data or knowledge of the chemical form(s) for each element.

[c] Not used in this project.

1. The radionuclide classification scheme of the Reactor Safety Study should be used in lieu of element-dependent data.

2. Before reactor vessel failure, releases of radionuclides from the core materials should be estimated using temperature-dependent release rates for representative species. These should be used in combination with time-dependent temperature profiles of the various regions of the core to obtain estimates of the fractions of various radionuclides released from the core.

3. After reactor vessel failure, releases of radionuclides from the core materials should be estimated using element-dependent vaporization loss fractions. All gases produced by concrete decomposition should be assumed to pass through the melt.

4. Leaching from the core materials should be estimated for limited-core-damage accidents using available experimental leach rates.

5. No retention in the primary coolant system should be assumed for equipment qualification purposes. Limited aerosol retention in the primary coolant system may be assumed for certain meltdown accidents for emergency planning.

6. All radioactive species other than noble gases, elemental halogens, and organic halides should be assumed to be present as aerosols in the containment.

7. The reduction of the radionuclides in the containment atmosphere by containment sprays, filter systems, ice-bed condensers, and other engineered safety features should be taken into account. In addition, the reduction of those radionuclides by natural processes may be considered; however, the effects of aerosol concentrations higher than those in the Containment Systems Experiments should not be considered.

8. Iodine should be taken to be present initially either as elemental iodine or as CsI, depending on which assumption would result in the largest doses to the members of the population of concern. The only exception is that 3.0% of the iodine should be assumed to be present initially as organic iodine.

9. Partitioning of noble gases and elemental iodine between the containment atmosphere and water in the containment should be included.

10. The effects of radioactive decay can be taken into account.

The following assumptions are also appropriate for estimating source terms for interim purposes:

1. Best estimates of the radionuclide releases for bounding sequences, modified by a safety factor, should be used for equipment qualification. All accidents considered should be within the design basis.

2. Best estimates of the radionuclide releases for a wide spectrum of accident sequences appropriate to the plant under consideration should be used for emergency planning. Accidents ranging from those involving only gap releases to those involving complete meltdown of the core should be considered.

3. All accidents should be treated "mechanistically," using plausible accident scenarios.

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4. EQUIPMENT QUALIFICATION

4.1 INTRODUCTION

Many of the safety-related components used in a nuclear plant are selected partially on their ability to withstand on a long-term basis the radiation fields which are encountered during normal operations and partially on their ability to withstand on a short-term basis the exceptional fields which might be encountered during accidents. This chapter illustrates the assumptions which are taken to be appropriate for the estimation of radionuclide source terms for such considerations, that is, for equipment qualification.

4.2 DESIGN BASIS ACCIDENT SPECTRUM

For the treatment of equipment qualification in this report, only accidents whose releases would bound the design basis envelope need to be considered. One such bounding accident* might involve melting of a large fraction of the core** as the result of delayed functioning of the emergency core cooling system following a large pipe break in the primary coolant system. Although the initial radioactive releases would encounter a relatively dry pathway to the containment, at least the releases toward the end of the accident would encounter water in that pathway.

Another possible bounding accident might involve melting of a large fraction of the core as the result of delayed functioning of the emergency core cooling system following a transient event.*** In a PWR, any radionuclides released from the core in such an accident might pass through water in the pressurizer quench tank on their way to the containment. In a BWR, the released radionuclides would typically bypass the drywell, be released through the relief valves and enter the suppression pool.

An important difference between the two accidents just described would be in the paths followed by the releases from the core materials. Whereas for the pipe-break accident the entire path to the containment often would be relatively dry until the emergency core cooling started to function, for the transient-initiated sequence the released materials might encounter water in that path throughout the accident. Thus a substantial fraction of the released radionuclides might be entrained by the water throughout the accident.

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*This description is based on the Technical Bases Report (Denning, 1981a).

**In general, if more than approximately 50-80% of the core melted, melting probably could not be terminated and the entire core would melt (USNRC, 1975). Accidents with complete meltdown of the reactor core are outside the design basis.

***This description is also based on the Technical Bases Report (Denning, 1981a).

Another potentially important difference between the two accidents would be in the residence times of the released radionuclides in the primary coolant system. Although the pipe-break accident might have relatively short coolant system residence times during much of the accident, the transient-initiated accident might have relatively long coolant system residence times during a large portion of the accident. Thus if aerosol releases were large, aerosol deposition processes within the coolant system might be much more important for the latter transient accident. This would, however, be very dependent on both the reactor design and the details of the accident scenarios.

To consider both accident sequences, the overheating of the core was assumed to proceed until temperatures sufficient to permit melting in 50% of the core were attained. (More exactly, temperatures sufficient to permit melting in 50% of the core regions, according to MARCH calculations, were assumed.) Then further overheating was taken to be prevented by activation of the emergency core cooling system. That system was assumed to fail completely until it started to function to cool the core adequately. For the loss-of-coolant accident (LOCA), a short unobstructed path through the primary coolant system was assumed. The overheating of the core was assumed to follow the same history for both these accident sequences.

4.3 RELEASES FROM THE CORE MATERIALS

The fractions of the core materials predicted to be released for both accident sequences investigated in the illustrative calculations are the same and are given in Table 4.1. Also given are the corresponding TID source terms (USAEC, 1962) currently used for many regulatory considerations. As can be seen in the calculations performed for this study, the releases for three important element groups (Cs-Rb, Te-Sb, and Ba-Sr) are estimated to be much larger than the TID source term values while the releases for one other group (La) are estimated to be much smaller than the TID values.

It is important to note, however, that these estimated amounts are highly dependent upon the exact accident scenario considered. For example, the sequence used in the illustrative calculations was a terminated AD* for a large containment PWR (the RSS PWR). This sequence is postulated to take place over a relatively short period of time. Other sequences which involve comparable total core damage but which involve slower overheating of the core might be expected to result in

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*See Appendix A for an explanation of this notation.

Table 4.1. Release fractions from core materials

Report	Release	Fractions released from core materials						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report[a,b]	Gap + "1/2" meltdown	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)[c]
	Leach	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
TID-14844	Total	1.00	1.00	0.01	0.01	0.01	0.01	0.01

[a] "Best" estimate.

[b] These vaues have been rounded off. The exact estimates are given in Appendix B of the other report for this project.

[c] 1(-4) denotes 1×10^{-4}.

somewhat larger releases of certain materials. According to the methods used here, somewhat larger releases of all the groups from Te-Sb on over shown in Table 4.1 would occur in a relatively slow sequence. In calculations of certain slow sequences with core cooling initiated after 50% of the core regions was melted, the estimated releases of the last four groups might easily be double the amounts given in Table 4.1. (See the detailed time-dependent release results presented in Appendix B of the other report for this project [Niemczyk and McDowell-Boyer, 1982].)

In addition, the definition of the maximum extent of melting must be noted to be somewhat arbitrary. A different definition than that used here would result in different releases of at least some radionuclides. Also the exact point in the scenario of any accident after which complete melting could not be prevented is not known.

In general, all the differences due to consideration of various accident sequences and those due to different accident definitions would not noticeably affect the estimated releases of the more volatile species, such as the noble gases, the halogens and the alkali metals (Cs-Rb). The differences would be most evident for radionuclides in all the other less volatile groups. Most importantly, the differences potentially caused by both considerations are well within the uncertainties of the methods used to obtain the source term estimates.

For bounding accidents such as those considered here, the additional fractions released by leaching usually would be relatively small compared to the fractions initially released by overheating (see Table 4.1). It should be noted that these fractions would be very accident dependent and that the rates of leaching are not well known.

4.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

The two general accident sequences considered admit an almost infinite variety of initial distributions of radionuclides between the containment atmosphere and the water in the containment. Radionuclides initially retained in the PCS by processes such as aerosol removal could be entrained in the coolant water when the core was reflooded. Thus most of the released radioactivity generally could enter the containment in one way or another. The only question often would be the distributions of the radionuclides between the air and the water.

The extreme initial distributions entering the containment are listed in Table 4.2. For comparison, the source terms currently used for regulation are also presented (USAEC, 1974). Those regulatory source terms are basically the TID ones, with the amounts of "solid" fission products being doubled for conservatism.

Table 4.2. Total release fractions for design basis accident for equipment qualification

Report	Initial distribution	Fractions of core inventory released to containment but not to environment						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report[a]	Airborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)[b]
	Waterborne[c]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
	Waterborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)
	Waterborne[c]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
Regulatory Guide 1.89	Airborne	1.00	0.25	0.00	0.00	0.00	0.00	0.00
	Waterborne	0.00	0.50	0.01	0.01	0.01	0.01	0.01
	Plateout	0.00	0.25	0.01	0.01	0.01	0.01	0.01

[a] "Best" estimates for the two extremes of initial distributions within the containment.

[b] 1(-4) denotes 1 × 10^{-4}.

[c] The values on this line represent the additional releases due to leaching to the core materials.

4.5 TRANSPORT WITHIN THE CONTAINMENT

For equipment qualification, the containment constitutes the environment of concern. Thus, the only important factor in the containment for source term considerations is the transfer across the interface between the containment atmosphere and any water in the containment. This is described by the consideration of partitioning given in Section 3.3 of Chapter 3 of the other report for this project (Niemczyk and McDowell-Boyer, 1982). All other considerations of movement within the containment are beyond the scope of this project and so are not considered here.

4.6 UNCERTAINTIES

The uncertainties associated with estimates of the releases from the core materials are dependent on the species involved. For example, for accidents involving melting of a large fraction of the core, essentially the entire amounts of the more volatile species generally would be released. The main uncertainty in predicting releases for many such species is associated with estimating the amounts of those species present in the core inventory both at the time of accident initiation and later in the accident. In contrast, for some species of low volatility, the fractions of the core inventory released are highly uncertain because the release rates for certain of those species are not well known. A lack of appropriate data and adequate models for describing releases contributes to that problem. Furthermore, there is again the problem of estimating the amounts present in the core inventory for such species. In the procedures used in this report, there is the additional uncertainty associated with using element groups instead of considering radionuclides on an element by element basis because there is often a wide range of release rates associated with the various elements within a given group. Furthermore, although the more volatile species would typically be entirely released in any accident involving melting of a large fraction of the core, the amounts of the less volatile elements releases would be highly scenario dependent. And even if the scenario is specified, its thermal-hydraulic description is somewhat uncertain and that affects the amounts of at least the less volatile species predicted to be released.

The uncertainties associated with the estimates of the releases from the primary coolant system are also dependent on the species involved. For example, noble gases would be entirely released from the PCS. In contrast, other species could be retained in varying amounts depending on their volatilities, their reactivities, and the accident scenario. However, because the design basis approach used for equipment qualification is a bounding approach, these uncertainties are not overly important if even just one accident can be envisioned which would probably result in very low retention of all species. Consequently, the most important uncertainties for estimating design basis source terms for equipment qualification are those associated with the factors which determine the initial releases of radionuclides from the core materials (see Table 4.3). (For more extensive discussions of the detailed sources of uncertainties, see Appendixes B-D of the other report for the project. For a related discussion of general sources of uncertainties, see Appendix E of that report.)

4.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EQUIPMENT QUALIFICATION

A reasonable procedure for estimating source terms for equipment qualification would be to use the detailed procedures and/or assumptions described in Chapter 3 to consider various accidents potentially bounding the design basis. Thus, best estimate design basis source terms appropriate for a given reactor could be estimated. However, because of the inherent complexity of some of the indicated procedures and assumptions, there is a significant likelihood that errors would be made in their application by persons not thoroughly knowledgeable in source term estimation. Thus, it might be more reasonable to use "generic" (i.e., for a typical reactor) source terms which had been generated by persons expert in the indicated procedures. Given the current level of sophistication of source term estimation, such a set of source terms would be relatively reactor independent and therefore could probably be used for a large number of reactors.

In any case, because of the many problems associated with the definition and the description of the bounding accident(s) for the design basis for any reactor, the worst-case accident(s) for any particular reactor cannot be described exactly. Thus, the best-estimate source terms for either a specific reactor or a generic reactor should be modified to account for some of the uncertainties.

In particular, because of the large uncertainties associated with certain aspects of the estimation of source terms for equipment qualification, it is suggested that the best estimates of sequences such as those considered here be multiplied by an adjustment factor. More exactly, the value of 1.3 is suggested for species of high volatility (noble gases, halogens, and alkali metals) and the value of 2.0 is suggested for all other species. Examples of such source terms are presented in Table 4.4.

For the more volatile species, which would be released entirely regardless of the details of the accident description, this adjustment factor encompasses the uncertainty associated with radionuclide core inventory estimates provided by such codes as ORIGEN (Bell, 1973). For the less volatile species, whose estimated releases would depend upon the conditions assumed, this would include some variations due to the arbitrary boundary conditions chosen. It would not account for the very large uncertainties associated with the release rates for some of those less volatile species. However, it might be noted that, in general, the release rates used in this study for those species with highly uncertain release rates have been taken to fall toward the high ends of their possible ranges.

Table 4.3 Impacts of problems in estimating magnitudes of source terms on equipment qualification accident sequences

Phenomena	Description	Problems[a]	Potential impact	
			Individual accident	Design basis envelope
Releases				
Releases other than leaching	Empirical temperature-dependent rates	Insufficient data	Large	Large
	Thermal-hydraulic estimates	Insufficient data; inadequate models		
Leaching releases	Empirical rates	Insufficient data	Small	Small
Transport Through Primary Coolant System				
Transport through coolant system atmosphere	"First-principles" transport code	Insufficient data; inadequate models	Large	Small
	Thermal-hydraulic estimates	Insufficient data; Inadequate models		
Scrubbing in coolant system	Empirical rates	Insufficient data; Inadequate models	Large	Small

[a] See Table 3.1 in Chapter 3 for more details.

Table 4.4. Source terms for equipment qualification

Initial distribution	Fractions of core inventory released to containment but not to environment						
	Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
Airborne	1.00	1.00	1.00	1.00	0.010	0.02	2(-4)[a]
Waterborne[b]	0.00	0.00	0.00	0.00	0.002	0.002	2(-4)
Waterborne	1.00	1.00	1.00	1.00	0.010	0.02	2(-4)
Waterborne[b]	0.00	0.00	0.00	0.00	0.002	0.002	2(-4)

[a] 2(-4) denotes 2×10^{-4}.

[b] The values on this line represent the additional releases due to leaching to the core materials.

References for Chapter 4

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5. EMERGENCY PLANNING

5.1 INTRODUCTION

The procedures which would be followed to mitigate the radiation hazard to the human population during a reactor accident would depend on the anticipated releases of radioactivity to the environment. This chapter illustrates the assumptions which are taken to be appropriate for the estimation of radionuclide source terms for such considerations, that is, for emergency planning.

5.2 ACCIDENT SPECTRUM

For emergency planning, a broad spectrum of possible accidents needs to be investigated. A representative variety of postulated accidents and accident sequences for a few nuclear plants are described in some detail in Appendix A of this report and in greater detail in the reports from which those descriptions have been adopted. In this chapter, for illustrative purposes, the following accidents are considered: a normally-terminated, large pipe loss-of-coolant accident (LOCA) involving gap releases only; a belatedly-terminated, large pipe LOCA involving melting of 50% of the core; and a variety of meltdown accidents. They are described briefly in the next several paragraphs.

For the normally-terminated LOCA, only minor overheating of the core would occur. Because the core would heat up rather unevenly in such an accident, releases beyond just the classic gap releases would occur for some fuel rods while not even the classic gap releases would occur for many other rods. Overheating would soon be terminated by reflooding of the core by the emergency core cooling system. Escape of radioactivity to the environment would be relatively slow, by leakage from the containment. For this study, it was assumed that overheating would occur only until 50% of the core was above the minimum cladding rupture temperature.* (More exactly, "gap" releases were defined to be the releases which would occur before 50% of the regions of the core reached, according to MARCH calculations, the temperature at which cladding would rupture.*)

For the accident involving delayed functioning of the emergency core cooling system, the overheating of the core would be much more extensive than for the first accident considered. Some of the core could experience temperatures sufficient to cause melting while others would experience only much lower temperatures. Eventually activation of the emergency core cooling system would reflood the core and prevent further melting. Escape of radioactivity from the containment would be by leakage only. For this study, it was assumed that overheating would occur until 50% of the core was above the melting temperature. (More exactly, temperatures sufficient to permit melting in 50% of the regions of the core, according to MARCH calculations, were assumed.) For both this accident and the normally-terminated LOCA, the pathway through the primary coolant system was taken to be short and unobstructed. In addition, leaching of the reflooded core materials was assumed to occur.

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*This artifice was adopted for this illustrative calculation because of a lack of readily available appropriate thermal-hydraulic results for a normally-terminated LOCA.

For any meltdown accident, more overheating of the core would occur than for either of the preceding two accidents. Any meltdown accident would involve melting of a large fraction of the core, as well as the potential for substantial interactions of at least part of the core with both the reactor vessel and the concrete basemat of the containment. A wide variety of accident conditions would be possible for meltdown accidents, ranging from those which would result in very small releases of radioactivity to the environment to some others which would result in relatively large releases. The specific sequences considered included some involving relatively rapid meltdown of the reactor core, as well as others involving relatively slow meltdown. In addition, the sequences investigated included some with the potential for substantial permanent retention of at least some radionuclides in the primary coolant system, in addition to other sequences with little likelihood of any permanent retention in that system. Furthermore, the sequences considered varied widely in both the modes and the timing of containment failure assumed, and included both sequences with early massive rupturing of the containment and concomitant flows of large masses of radioactivity from the plant, as well as some with only slow leakage of small amounts of radioactivity from the plant. Descriptions of some of the specific individual meltdown sequences investigated are given in Appendix A.

5.3 RELEASES FROM CORE MATERIALS

Accidents in the overall spectrum for emergency planning range from those involving releases of essentially negligible fractions of all radionuclides from the core materials to those involving releases of substantial fractions of many radionuclides. The fractions predicted to be released from the core materials in all the illustrative calculations performed for this study are given in Table 5.1. For consideration of releases during limited-core-damage accidents, the accidents investigated were the two described in the previous section. For consideration of releases during meltdown accidents, the accidents investigated were seven different sequences in the two Reactor Safety Study (RSS) plants* (AD, V, S_2D, and TMLB′ in the RSS PWR and TC, TW, and TQUV in the RSS BWR).** (The detailed results for all the sequences

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*The two "generic" reactors considered in the Reactor Safety Study were a large dry containment PWR (Surry) and a Mark I BWR (Peach Bottom).

**See Appendix A for descriptions of these sequences.

Table 5.1. Release fractions from core materials

Report	Release	Fractions of core inventory released from core materials[a]						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report	"gap"	0.04(0.96)[b]	0.01	0.01	0.001	1(-4)[c]	1(-6)	1(-7)
RSS	"gap"	0.03	0.017	0.05	1(-4)	1(-6)	0	0
This report	gap + "1/2" meltdown	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)
RSS	gap + "1/2" meltdown	-	-	-	-	-	-	-
This report	gap + meltdown[d,e]	1.00	1.00	1.00	1.00(0.78-1.00)	0.35(0.10-0.50)[f] 0.15(0.04-0.23)	0.02(0.003-0.02)[g] 0.15(0.06-0.25)	0.001[2(-4)-0.002]
This report	gap + meltdown[h]	1.00	1.00	1.00	0.80(0.57-1.00)	0.10(0.06-0.46) 0.04(0.03-0.21)	0.004(0.002-0.02) 0.05(0.02-0.20)	0.001
RSS	gap + meltdown	0.90	0.90	0.81	0.15	0.10	0.03	0.003
This report	vaporization[e]	0.00	0.00	0.00	0.00(0.00-0.22)	0.04(0.02-0.04)[f] 0.00	0.02(0.02-0.03)[g] 0.12(0.08-0.16)	0.01
RSS	vaporization	0.10	0.10	0.19	0.85	0.01	0.05	0.010
This report	total melt[d,i]	1.00	1.00	1.00	1.00	0.39(0.14-0.52)[f] 0.15(0.04-0.23)	0.04(0.02-0.04)[g] 0.27(0.21-0.37)	0.01
This report	total melt[h]	1.00	1.00	1.00	1.00	0.14(0.10-0.48) 0.04(0.03-0.21)	0.02(0.02-0.04) 0.18(0.16-0.32)	0.01
RSS	total melt[e]	1.00	1.00	1.00	1.00	0.11	0.08	0.013

[a] Value given denotes "representative" value for sequences considered; values in parentheses denote calculated ranges for those sequences.

[b] Denotes additional release by leaching.

[c] 1(-4) denotes 1 × 10^{-4}.

[d] These values assume no decrease in the fractional release rates after slumping of the core into the bottom of the reactor vessel.

[e] Based on RSS terminology.

[f] Upper line is based on barium data; lower line is based on strontium data.

[g] Upper line is based on ruthenium data; lower line is based on molybdenum data.

[h] These values assume a decrease in the fractional release rates after slumping of the core into the bottom of the reactor vessel.

[i] Includes gap + meltdown + vaporization.

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considered are presented in Appendix B of the other report for this project [Niemczyk and McDowell-Boyer, 1982].) The corresponding RSS source terms sometimes used for certain emergency planning purposes are also given in Table 5.1 for comparison (USNRC, 1975).

The releases estimated for the accident involving only "gap" releases depend on the details of the accident description adopted. For example, the estimates of the "gap" releases can vary substantially, depending on the rupture temperature assumed for the fuel rod cladding. In all cases, however, those estimated releases are relatively small. For a cladding failure temperature of 750°C, the estimated releases are essentially just the classic gap releases for half of the core (see Table 5.1).*

The releases estimated for the accident involving partial melting of the core also depend upon the particulars of the postulated scenario. The releases for such an accident were discussed in the previous chapter and so they are not discussed further. They are given in Table 5.1.

The radionuclide releases for accidents involving complete melting of the core likewise depend strongly upon the accident descriptions. From the ranges of the radionuclide releases for the seven meltdown accident sequences given in Table 5.1, it can be seen that for the first four groups of elements in the table (Xe-Kr, I-Br, Cs-Rb, and Te-Sb), the total amounts of the radionuclides predicted to be released are the same as in the RSS. However, as can also be seen in the table, the timing of those releases from the core materials with respect to before and after reactor vessel failure is much different in this study and in the RSS for some of those species. For each of the next two groups in the table (Ba-Sr and Ru), both the magnitudes and the timing of the releases are somewhat different than in the RSS. For the last group in the table (La), both the magnitude and the timing are approximately the same as in the RSS. All of the differences in the estimates of the releases from the core materials in this study and in the RSS are due to both the much different approaches used for estimating releases in the two studies and the use of some recent data in this study.

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*According to "best-estimate" calculations for a normally terminated large pipe LOCA (Johnson, Childs and Broughton), the conditions estimated in this study to be present when 50% of the core has reached 750°C are approximately equivalent to, or slightly worse than those expected for such a LOCA. Thus the choice of 750°C seems reasonable to describe a "gap" release, given the other large uncertainties incurred in describing such a release.

The conditions described here have been chosen on the basis of availability of MARCH output to use for these estimates. More appropriately, the output of computer codes better suited than MARCH to describe terminated LOCA's should be employed to consider "gap" releases. (See the brief discussion in Section C.3 in Appendix C which compares MARCH and more appropriate codes such as RELAP [Aerojet Nuclear Company, 1976].) The use of such codes would obviate the need for an artificial definition of gap releases such as that used in this study.

In general, because of the use of temperature-dependent release rates in this study, the total estimates releases of the less volatile species depend strongly on the length of time any overheating occurs, so that accidents involving slower heating often result in larger releases of many of those species. This accounts for most of the variation seen in Table 5.1 for different meltdown sequences. The other major source of variation for the meltdown sequences in that table is due to the two alternate assumptions employed to describe the releases after slumping of the core materials into the bottom of the reactor vessel. As can be seen in that table, only the estimates for the Ba-Sr group and the Ru group are very dependent on whether the release rates are assumed to remain the same or to decrease after slumping.

5.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

The continuum of accidents which should be considered for emergency planning includes an essentially infinite variety of possiblities with respect to transport through the primary coolant system (PCS). Accidents ranging from those involving almost complete retention of some released radionuclides in the PCS to those involving retention of none of the released radionuclides are possible. In addition, transport through the PCS could affect whether the radionuclides would escape to the containment as airborne species or as waterborne ones.

For the illustrative calculations, the escape fractions from the PCS were taken, based on the considerations reviewed in Chapter 3, to be those given in Tables 5.2 and 5.3. (These fractions represent the greatest retention which might currently be considered.) These escape fractions apply only to the amounts of materials entering the containment via the primary coolant system. Typically, in previous calculations such as those performed for the RSS, no retention in the PCS has been assumed. (The one exception in the RSS in which retention in the PCS was considered was a meltdown accident in a BWR. See the discussion in Appendix C of the other report for this project.)

As can be seen, the assumptions reviewed in Chapter 3 in general do not result in large permanent retention in the PCS for most accidents. They can, however, result in a redistribution of a sizeable fraction of some species between the PCS atmosphere and any water in the PCS. In addition, for a few sequences, they can result in significant permanent retention for some radionuclides groups which are both transported as aerosols and released mostly before vessel failure, e.g., Cs-Rb and Te-Sb.

Table 5.2 Summary of primary coolant system escape fractions

Type of accident	Escape fractions		
	Xe	I[a,b]	Aerosols[a]
A. Accident sequence without water in the pathway through the PCS			
Large LOCA's	1.0	1.0	1.0 (0.8-1.0)
Small LOCA's	1.0	1.0	1.0 (0.3-1.0)
Transient-initiated LOCA's	1.0	1.0	1.0 (0.3-1.0)
Other transients	1.0	1.0	-[c] (0.1-1.0)
Accident involving "gap" release only	1.0	1.0	1.0 (0.1-1.0)
Accidents involving partial melting	1.0	1.0	1.0 (0.1-1.0)
B. Accident sequences with water in the pathway through the PCS[d]			
Meltdowns	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)
Accidents involving "gap" release only	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)
Accidents involving partial melting	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)

[a] The values given in parentheses denote ranges of reasonable values for escape fractions.

[b] Applies to iodine present as elemental iodine.

[c] The escape fractions for these sequences are highly dependent on the rate of flow of steam through the PCS during the accident. See Table 5.3 for the details.

[d] Obviously, the amount of retention in the PCS water would depend upon the path taken by the radionuclides, that is, upon the amount and temperature of the water encountered.

Table 5.3 Primary coolant system escape fractions for transient-initiated meltdowns

Accident sequence	Escape fractions		
	Xe	I[a]	Aerosols[b]
TMLB′ (PWR)[c]	1.0	1.0	1.0 (0.8-1.0)
TC (BWR)	1.0	1.0	0.33 (0.10-0.33)
TW (BWR)	1.0	1.0	0.33 (0.10-0.33)
TQUV (BWR)	1.0	1.0	0.67 (0.33-1.00)

[a] Applies to iodine present as elemental iodine.

[b] The values given in parentheses denote ranges of reasonable values for escape fractions.

[c] The values given here do not include the effects of possible scrubbing by water in the pressurizer quench tank (see Table 5.2).

5.5 TRANSPORT WITHIN THE CONTAINMENT

The total fractions of the various radionuclides estimated to be released from the containment in some representative accident sequences for three different plants considered in the Reactor Safety Study and in the Reactor Safety Study Methodology Applications Program (RSSMAP; Carlson et al., 1981; Kolb et al., 1981; Hatch, Cybulskis, and Wooton, 1981; Cybulksis, 1981; and Wooton, 1981) are listed in Table 5.4. (The estimated releases for many more sequences are given in the addendum of Appendix E of the other report for this project.)

As can be seen by the comparisons presented in Table 5.4 (and the tables in Appendix E of the other report), the values estimated in this study and in the RSS and in the RSSMAP follow-on work are comparable except for the following situations: (1) transient-initiated meltdowns with scrubbing of radionuclides assumed in the pressurizer quench tank (e.g., TMLB'); and (2) accidents with substantial aerosol removal assumed in the primary coolant system (e.g., TC). In both types of accidents, the source terms estimated in this report tend to be somewhat lower than those in the RSS and in the RSSMAP. In contrast, for all accidents, the Ba-Sr group releases and the Ru group releases are generally larger in this report if the potential decreases in the initial release rates after slumping into the bottom of the reactor vessel are ignored; however, if such decreases are assumed, then the releases of both the Ba-Sr group and the Ru group are typically lower in this report. (For example, see the values given in Table 5.4 for event V.) It should be noted that the differences indicated for the two element groups are well within the uncertainties associated with the descriptions of source terms.

5.6 UNCERTAINTIES*

For a given accident sequence, one of the largest sources of uncertainty is in the basic description of the associated accident scenario(s). In particular, even though the likelihood of conditions suitable for the occurrence of certain events can be predicted, the likelihood of the events themselves often cannot be. For example, the computer codes used to describe the environment in the containment can be utilized to predict the presence of conditions which might result in failure of the containment but the actual mode and timing of containment failure during any accident cannot be predicted and must be assumed. Likewise the accident descriptions abound with many other examples of phenomena which are assumed and not predicted. Consequently, it must be realized that the description of any given accident sequence involves many sometimes rather arbitrary assumptions and that if some of those assumptions were changed, the description of the accident might likewise change significantly. Because an accident spectrum is

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*A more complete description of many types of uncertainties is given in Appendix E of the other report for this project.

Table 5.4. Containment escape fractions for some representative accident sequences

Report	Sequence	Reactor	Cumulative fracture of core inventory released to environment[a]						
			Xe-Kr	I-Br[b]	Cs-Rb	Te-Sb	Ba-Sr[c]	Ru[d]	La
This report	V	RSS PWR	1.00	0.70	0.94	0.94	0.20(0.47)	0.025(0.21)	0.004
This report	V[e]	RSS PWR	1.00	0.70	0.94	0.80	0.04(0.11)	0.01(0.09)	0.004
RSS[f]	V	RSS PWR	1.00	0.64	0.82	0.41	0.10	0.04	0.006
This report	TMLB′-ε	RSS PWR	0.86	0.003	0.001	0.001	0.0002(0.0004)	3(-5)(0.0003)	1(-5)[g]
This report	TMLB′-ε[h]	RSS PWR	0.86	8(-4)	3(-4)	3(-4)	6(-5)(0.0001)	2(-5)(0.0001)	1(-5)
RSS[f]	TMLB′-ε	RSS PWR	0.70	4(-4)	0.001	0.001	1(-4)	1(-5)	1(-5)
This report	TC-γ′	RSS BWR	1.00	0.47	0.67	0.67	0.04(0.14)	0.02(0.15)	0.006
This report	TC-γ′[i]	RSS BWR	1.00	0.47	0.22	0.26	0.013(0.064)	0.02(0.11)	0.006
RSS[f]	TC-γ′	RSS BWR	1.00	0.45	0.67	0.64	0.07	0.05	0.008
This report	TQUV-γ	RSS BWR	1.00	0.02	0.05	0.07	0.002(0.010)	0.003(0.02)	0.001
This report	TQUV-γ[i]	RSS BWR	1.00	0.02	0.03	0.05	0.002(0.009)	0.003(0.020)	0.001
RSS[f]	TQUV-γ	RSS BWR	1.00	0.02	0.06	0.11	0.006	0.007	0.001
This report	S_2HF-γ	IC PWR[j]	1.00	0.13	0.63	0.63	0.06(0.16)	0.02(0.13)	0.005
RSSMAP	S_2HF-γ	IC PWR	1.00	0.13	0.57	0.49	0.07	0.04	0.007
This report	AD-δ	IC PWR	1.00	7(-7)	3(-7)	3(-7)	6(-8)[2(-7]	6(-9)[6(-8)]	6(-10)
RSSMAP	AD-δ	IC PWR	1.00	7(-7)	3(-7)	3(-7)	3(-8)	2(-8)	4(-9)
This report	A	RSS PWR	4(-6)	2(-7)	1(-7)	1(-8)	1(-10)[5(-10)]	1(-11)[5(-10)]	1(-12)
RSS	A	RSS PWR	3(-6)	1(-7)	6(-7)	1(-9)	1(-11)	0	0

[a] For other sequences, see Appendix E of the other report for this project. Also see Appendix E of that report for details.

[b] Represents fraction of iodine present at elemental iodine which escapes.

[c] Based on strontium data; value in parentheses based on barium.

[d] Based on ruthenium data; value in parentheses based on molybdenum.

[e] Release rates after slumping reduced by surface-to-volume considerations.

[f] Based on the RSS-rebaselining work in the Reactor Safety Study Methodology Applications Program (RSSMAP).

[g] 1(-5) denotes 1 × 10^{-5}.

[h] Removal due to scrubbing by passage through pressurizer quench tank included.

[i] Aerosol retention in primary coolant system included.

[j] Denotes ice condenser PWR considered in the RSSMAP.

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composed of many accident sequences whose basic descriptions are highly uncertain, various aspects of the spectrum are also uncertain.

The uncertainties associated with estimates of the releases from the core materials and retention in the primary coolant system are dependent on essentially the same factors already described in Section 4.5 for equipment qualification. The main difference is that because of the different approaches used for equipment qualification and emergency planning, the uncertainties in the retention in the primary coolant system are more important for emergency planning. Although for many sequences with little retention anticipated, the uncertainties associated with the fractions escaping from the PCS would be relatively low for most species, for some other sequences with significant retention possible the uncertainties would be relatively high for many species. In addition, such retention of some species in certain sequences would be dependent on the details of the comparatively uncertain thermal-hydraulic conditions.

Like the uncertainties for both release from the core materials and escape from the coolant system, the uncertainties associated with the estimates of escape from the containment are also dependent on the species, with the uncertainties for noble gases being smallest. In addition, the uncertanties depend on the accident scenario. For example, for accidents involving early containment failure the uncertainties associated with transport within the containment are similar for most species whereas for accidents involving either delayed failure or no failure of the containment, the uncertainties are different and depend upon the possible fates of the various species. Except for iodine, these detailed fates are not generally addressed and so the uncertainties are large for some species. The largest uncertainties of importance are those associated with the mode and timing of containment failure. In a meltdown accident if the timing of the failure of the containment is changed somewhat, the releases of radioactive materials to the environment can change substantially (see Table 5.5).

5.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EMERGENCY PLANNING

To estimate source terms for emergency planning, the assumptions and procedures outlined in Chapter 3 of this report might be implemented to generate source terms for an appropriate spectrum of accidents for the plant under consideration. Unfortunately, such an undertaking is a difficult process and includes the potential for many problems. Thus a more appropriate procedure might be to use "generic" source terms for various classes of accidents (Pasedag, Blond, and Jankowski, 1981).

For example, on the basis of the estimated magnitudes of the radionuclide releases to the environment for all the sequences considered in this project, all accident sequences can be divided into several classes. As one possibility, the six PWR classes and the five BWR classes listed in Table 5.6 might be considered. Alternatively,

Table 5.5 Impacts of problems in estimating magnitudes of source terms on emergency planning accident sequences

Phenomena	Description[a]	Problems[a]	Potential impact	
			Lesser-core damage accidents	Meltdown accidents
Releases				
Releases before vessel failure other than leaching	Empirical temperature-dependent rates	Insufficient data	Large	Large[b]
	Thermal-hydraulic estimates	Insufficient data; inadequate models		
Leaching releases	Empirical rates	Insufficient data	Small	Small
Releases after vessel failure	Empirical temperature-dependent rates	Insufficient data; inadequate model	-	Large
Transport Through Primary Coolant System				
Transport through coolant system atmosphere	"First-principles" transport code	Insufficient data; inadequate models	Large	Large
	Thermal-hydraulic estimates	Insufficient data; Inadequate models		
Scrubbing in coolant system	Empirical rates	Insufficient data; Inadequate models	Large	Large
Transport through containment				
Transport in containment	Empirically-based code	Insufficient data; inadequate models	Small	Large
	Thermal-hydraulic estimates	Insufficient data; inadequate models		
Leakage or other release from containment	Empirically-based rates	Inadequate models for failure modes	-	Large

[a] See Table 3.1 in Chapter 3 for more details.

[b] For some radionuclides.

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Table 5.6 Estimated values of updated total containment escape fractions

Class	Corresponding RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr[b]	I-Br[c]	Cs-Rb	Te-Sb	Ba-Sr[d]	Ru[e]	La
PWR I′	PWR 2 (V only)	1.0	0.8	0.9	0.9	0.5;0.2	0.03;0.2	0.005
PWR I[f]	PWR 1	-	-	-	-	-	-	-
PWR II	PWR 2 + PWR 3	1.0	0.3	0.6	0.6	0.5;0.2	0.04;0.3	0.01
PWR III	PWR 4 + PWR 5	1.0	0.01	0.01	0.01	0.01;0.004	0.003;0.03	0.002
PWR IV	PWR 6 + PWR 7	1.0	0.003	0.001	0.001	4(-4)[g];2(-4)	3(-5);3(-4)	1(-5)
PWR V	PWR 8 + PWR 9	1.0	5(-4)	0.001	3(-4)	2(-5);1(-5)	1(-6);1(-5)	3(-8)
BWR I′	BWR 2	1.0	0.6	0.8	0.7	0.1;0.05	0.01;0.1	0.003
BWR I	BWR 1	-	-	-	-	-	-	-
BWR II	BWR 2 + BWR 3	1.0	0.5	0.2	0.3	0.1;0.06	0.02;0.1	0.007
BWR III	BWR 4	1.0	0.001	0.005	0.006	0.003;5(-4)	4(-4);0.003	2(-4)
BWR IV	BWR 5	1.0	3(-10)	1(-8)	2(-9)	2(-10);1(-10)	1(-11);1(-10)	3(-13)

[a] The values given here for each class are not necessarily all-inclusive; instead they are representative of those sequences explicitly considered in this report.

[b] Ignores decay.

[c] Represents fraction of iodine present as elemental iodine which escapes.

[d] First value based on barium data; second value based on strontium data.

[e] First value based on ruthenium data; second value based on molybdenum data.

[f] Steam explosions were not considered in this report.

[g] 4(-4) denote 4 × 10^{-4}.

four composite classes, such as those summarized in Table 5.7, might be formed.

Although the accident sequences of interest vary greatly among different plant designs, sequences which fall within each of the accident sequence classes have been found at each type of plant considered to date. (See the addendum in Appendix E of the other report for this project.) As a result, although the probabilities of occurrence of each class probably vary widely among plants (see Table E.4 of Appendix E of the other report), the magnitudes of the potential releases vary over the same basic ranges.

In general, most aspects of source term estimation which are currently thought to be potentially conservative and which might yield reductions of the estimated releases as the result of work in the near future will not result in reductions of the releases for all accident sequences. Therefore, in a sense, such improvements to a large extent may just change the relative probabilities of the various classes of accidents and not the anticipated releases for the classes.

Inasmuch as source terms for emergency planning would often be used in a probabilistic framework, best estimates such as those obtained in the illustrative calculations presented in the chapter are indicated. As a result, it is probably reasonable to adopt accident classes and associated source terms such as those presented in Table 5.7 for regulatory guidance for emergency planning.

Table 5.7 Estimated values of updated total containment escape fractions for generic accident classes

Class	Corresponding RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr[b]	I-Br[c]	Cs-Rb	Te-Sb	Ba-Sr[d]	Ru[e]	La
1	PWR 1 + PWR 2 + PWR 3; BWR 1 + BWR 2	1.0	0.8	0.9	0.9	0.5;0.2	0.03;0.2	0.005
2	PWR 4 + PWR 5; BWR 4	1.0	0.01	0.01	0.01	0.01;0.004	0.003;0.03	0.002
3	PWR 6 + PWR 7	1.0	0.003	0.001	0.001	4(-4)[f];2(-4)	3(-5);3(-4)	1(-5)
4	PWR 8 + PWR 9 BWR 5	1.0	5(-4)	0.001	3(-4)	2(-5);1(-5)	1(-6);1(-5)	3(-8)

[a] The values given here for each class are not necessarily all-inclusive; instead they are representative of those sequences explicitly considered in this report.

[b] Ignores decay.

[c] Represents fraction of iodine present as elemental iodine which escapes.

[d] First value based on barium data; second value based on strontium data.

[e] First value based on ruthenium data; second value based on molybdenum data.

[f] Steam explosions were not considered in this report.

[g] 4(-4) denote 4 × 10^{-4}.

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6. LIMITATIONS AND RECOMMENDATIONS

6.1 LIMITATIONS

6.1.1 Assumptions and Procedures

The project reported in this document was to use the information provided in the Technical Bases Report (USNRC, 1981) as the starting point for formulating assumptions appropriate for use in regulatory guidance. Regretfully, the information presented in the Technical Bases Report was incomplete in some respects. Because the study reported here was of such limited scale, it was not possible to address many of the previously undertreated topics at a greater level of detail than had been done before. Thus this report necessarily suffers from many of the same problems as the Technical Bases Report.

Among the acknowledged shortcomings of the work performed for the Technical Bases Report (Sherry et al., 1981) are the following:

1. Systematic analysis of fission product transport from the fuel to the environment was not performed for that study. Thus the effects of certain important interdependencies in the overall problem of source term estimation were ignored.

2. No detailed analysis of the uncertainties was made. Unfortunately, some of these uncertainties are large.

3. In many cases only a cursory examination of the transport behavior of potentially important fission product species (e.g., Te, Ru, Sr) other than cesium and iodine was made.

4. Only a limited number of accident sequences were evaluated so that the full range of possible accident conditions may not have been adequately covered.

5. A number of physical processes which may have the potential to significantly affect fission behavior were not evaluated in depth. For example, the effects of hydrogen combustion in the containment and the effects of attenuation of aerosols along leak paths were not addressed.

6. For the most part, fission product behavior during actual previous reactor accidents was not analyzed.

7. Fission product release and transport during accident sequences in which steam explosions are postulated to occur were not addressed.

To a large extent, many of the shortcomings of the Technical Bases Report merely reflect the lack of appropriate data and/or understanding which currently exists in certain areas. For example, at least the following potentially important areas have been understudied experimentally in the past: (1) the chemistry of almost all elements in accident environments; (2) the effects of hydrogen burning; (3) the release rates from core materials and/or corium for many elements; (4) the retention of vapors and aerosols in the primary coolant system; (5) the effects of condensing steam environments on aerosol behavior; (6) the effects of core-concrete interactions; (7) aerosol formation and behavior in LWR accident conditions; and (8) the removal of radionuclides by suppression pools and ice-bed condensers. (The needs in many of these areas are discussed in more detail in the Technical Bases Report [Sherry et al., 1981] and so are not considered here.) Furthermore, most of these areas have not been adequately treated theoretically Thus, many of the shortcomings of the Technical Bases Report are currently "necessary." As a result, these problems are of necessity not treated in detail in this report.

For the purpose of using the information provided in the Technical Bases Report in this project, there are some problems other than those already mentioned. Among those additional problems are the following:

1. The overall attitude of that document was in many respects a research oriented one, with much of the work not being directly applicable to regulation. For example, although some very interesting calculations were performed to investigate potential radionuclide retention in the primary coolant system, that work is not currently usable for regulatory considerations.

2. The rushed nature of the work performed for that study is apparent throughout the report. For example, there are certain inconsistencies and some potentially treatable areas not addressed in the report.

3. Variations among reactors were not adequately addressed in many portions of that document. This was especially the situation for the consideration of releases from the core and structural materials and transport through the primary coolant system.

4. Reasonable variations among possible accident conditions were not always addressed. For example, the effect of high pressures in the reactor vessel on releases from the core and structural materials was not considered.

5. The document incorrectly indicated that a consensus exists in certain technical areas where indeed no consensus of the experts does exist. Because definitive answers do not exist in many areas of source term estimation, the extent of disagreement among the experts

must be regarded as one very important indicator of the uncertainty associated with various aspects of source term estimation.

6. No attempt was made to put the accident descriptions in any appropriate perspective. Namely, little consideration of the strong dependence of individual accident descriptions upon many somewhat arbitrary assumptions.

In addition to the limitations associated with the Technical Bases Report and the information behind it, there are other limitations of the assumptions and procedures adopted here. Many of them result basically from the inherent complexity of source term estimation. For example, because some of the state-of-the-art procedures for estimating source terms involve the use of complicated models and/or intricate computer codes, there is much room for error in the application of the indicated procedures. It is far beyond the scope of this project to describe all the potential pitfalls associated with the procedures. Furthermore, other limitations exist because the models and codes themselves include many sources of error, such as unmodeled processes, mismodeled processes, and so forth, and so they are not necessarily applicable for all or even any accidents. Other potential problems arise because the models and codes which represent the "best" methods were not developed with regulatory uses in mind. Consequently, because both the models and codes and the use of the models and codes are associated with large uncertainties, the general application of the adopted methods for source term estimation for regulatory guidance must be acknowledged to be fraught with many potential problems.

A further limitation associated with the work presented in this report is the consideration of only accidents within the current design basis for equipment qualification.* Such restricted consideration ignores the potential impact of more severe accidents such as meltdowns. In addition, it essentially precludes any useful consideration of the time-dependent character of the radionuclide releases for all accidents.

6.1.2 Illustrative Estimates

The approach taken in this document is built on the work of many others. While better methods and procedures than some of those used here probably can and will be developed, they are not presently available. In addition, although data better than much of that now available probably can and are being obtained, they are not currently available. Thus the source term estimates presented here in some sense represent the state-of-the-art. However, the quality of the estimates in this report necessarily has been restricted by the limited scale of this project. In particular, there has not been time to straighten out or resolve certain inconsistencies and differences of opinion found in previous work and in the technical community. Also, previously performed calculations not initially intended to be used for the purposes of this document have been utilized as the bases for the estimates. In addition to this, there are all the uncertainties associated with any assessment of source terms.

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*The restriction of the accidents considered for equipment qualification to just those within the current design basis was part of the definition of the scope of this project by the funding agency.

6.2 RECOMMENDATIONS

As the result of the insights gained during this study, the following recommendations are made for future efforts in source term work:

1. The experimental research areas mentioned in the discussion of understudied areas should be investigated. In addition, improved models for releases of radionuclides and aerosols and transport of materials through the primary coolant system are indicated.

2. For some problems, a more scientific approach is indicated. For example, the fitting of curves to unrelated sets of data is often not a meaningful approach.

3. The sensitivities and uncertainties of source term estimation should receive far more consideration than they have received in the past. In particular, something beyond mere parametric analyses of computer codes is indicated. Much more effort should be expended to address the uncertainties associated with the problems themselves and not just those associated with the related models and computer codes.

4. Interaction among the people involved in various aspects of research and other investigations related to source term estimation should be fostered. In particular, more effort should be made to promote exchange of knowledge between chemists and nuclear engineers. It is clear that some chemists participating in research relevant to source term work are relatively ignorant of certain basic aspects of nuclear engineering which are important to source term estimation. Likewise, some nuclear engineers are comparatively unversed in the basic chemical factors which are important to source term estimation. In addition, more cooperation between workers at different laboratories involved in source term work would be beneficial.

5. An attempt should be made to incorporate much needed new perspectives and updated approaches into all aspects of source term work.

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APPENDIX A

ACCIDENT DESCRIPTIONS AND TERMINOLOGY

A.1 INTRODUCTION

Accidents involving core damage in light water reactors* can be divided into two basic groups: those accidents resulting in limited core damage (less than complete melting of the core) and those accidents resulting in complete meltdown. Examples of both of these groups are described in this appendix.

A.2 LIMITED-CORE-DAMAGE ACCIDENT DESCRIPTIONS

Limited-core-damage accidents are considered in this report for both equipment qualification and emergency planning purposes. On the basis of performance of the engineered safety features, two basic classes of limited-core-damage accident sequences can be defined: a class of accident sequences in which the engineered safety features function as anticipated after some accident initiating event, for example, a coolant system pipe break or a transient event, and another class in which one or more of the required engineered safety features function in a degraded mode after some accident initiator. On the basis of the characteristics of radionuclide transport in the primary coolant system, each of these classes can be further divided into two subclasses: one in which any released radionuclides must travel through water to reach the containment and another in which any released radionuclides do not pass through water to reach the containment during most of the accident. Examples of accidents in each of these classes and subclasses are described in the following paragraphs.

A.2.1 Accidents With Expected Functioning of Engineered Safety Features

Rod ejection. The ejection of a control rod from the reactor core in a PWR (or the dropping of a control rod in a BWR) would result in overheating of only a very localized region of the core. If the reactor protection systems functioned as intended, the transient produced by rod ejection would be countered by functioning of the reactor shutdown system. Likewise, if the engineered safety features functioned appropriately, the primary coolant system rupture (in the head of the reactor vessel caused by the rod ejection) would be compensated for by activation of the emergency core cooling system. Typically, the only radionuclide releases would be "gap" releases from a small region of the core, although, in some cases, much more extensive releases resulting from damage up to and including catastrophic rupturing of fuel rods in the vicinity of the ejected control rod could occur. Because the core would be surrounded by water, any radionuclides released from the core would directly enter the water in the primary coolant system.

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*Descriptions of the two basic types of light water reactors, namely, pressurized water reactors (PWR's) and boiling water reactors (BWR's), are given in the Technical Bases Report (Denning, 1981a). Descriptions of the variations of these two basic types of reactors which are considered in this report are also given in the Technical Bases Report (Denning, 1981a), as well as in the Reactor Safety Study Methodology Applications Program (RSSMAP) reports (Carlson et al., 1981; Hatch, Cybulskis and Wooton, 1981; and Kolb et al., 1981).

Main steam line break. In the event of a large pipe break in a main steam line in a PWR, steam would be lost rapidly, at least initially, from the steam line. The increased steam flow would cause increased heat removal from the primary coolant system and therefore decreased temperatures and pressures of the coolant in the primary system. If the engineered safety features operated as expected, any overheating of the core would be rather restricted. In general, the releases of radioactivity from the core would be very small; the only possible radionuclide releases would be "gap" and leaching releases from a few fuel rods located throughout the core. Any radionuclides released would enter the primary coolant system water directly. Unless a leakage path from the primary coolant system existed or developed, the released radioactivity would remain in that system.

Primary coolant system pipe break. In the event of a large pipe break in the primary coolant system, water would be lost rapidly from the core region. If the engineered safety features operated as expected, the overheating of the core before the emergency core cooling system reflooded the core region would be small. The overheating would generally be much more extensive than for either of the previously mentioned two accidents. "Gap" releases might occur for a large fraction of the core, although typically they would occur only for a small fraction of core. In addition, leaching of ruptured fuel rods might occur after the core were reflooded. In contrast to the situation for a large pipe break, a small pipe break would result in a slower rate of water loss from the core region. If the emergency core cooling functioned as expected, the core would not be uncovered and overheating would be minimal. Thus essentially no radioactivity would be released from the core.*

A.2.2 Accidents With Degraded Functioning of Engineered Safety Features**

Delayed functioning of the emergency core cooling system given a large pipe break. In the event of a large pipe break in the primary coolant system, coolant would be rapidly lost from the reactor vessel and the core would become uncovered. If the emergency core cooling system were to fail to deliver water to the reactor coolant system, the core would heat up and start to melt. The extent of melting, and therefore the amounts of various radionuclides released, would depend upon the length of the delay before the emergency core cooling system started to operate. The location of the break would have a substantial effect on the conditions encountered by radionuclides as they were transported through the reactor coolant system.* Although the initial releases would often encounter a relatively dry pathway to the containment, at least the releases toward the end of the accident would encounter water in that pathway.

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*This description of a small pipe break depends upon the operator recognizing the existence of the break and activating the engineered safety features before the break is detected by safety instrumentation.

**The descriptions in this subsection were adopted from NUREG-0772 (Denning, 1981a).

Delayed functioning of the emergency core cooling system given a transient event. In the event of a transient with loss of the normal heat sinks, water would eventually be boiled off the core and lost through the relief valves. If the emergency core cooling system were to fail to deliver water to the reactor coolant system, the core would begin to melt. Melting would be terminated if the emergency core cooling system started to operate before too much of the core were melted and complete meltdown of the core became inevitable. In a PWR, any radionuclides released from the core might pass through water in the pressurizer quench tank on their way to the containment. In a BWR, the released radionuclides would typically bypass the drywell, be released through the relief valves and enter the suppression pool. In both cases, the amounts of many radionuclides ultimately reaching the containment (or reactor building) atmosphere and the environment might be much smaller than the total amounts released from the core.

A.3 MELTDOWN ACCIDENT DESCRIPTIONS

Meltdown accidents are considered in this report only for emergency planning purposes.** Some of the meltdown accident sequences thought to dominate the risk to the human population for various categories of accident sequences are described here for each of the five different types of reactors investigated in this study. The descriptions given here have mostly been excerpted and/or adapted from the indicated source documents.*** [More detailed descriptions of all the accident sequences presented here, in addition to descriptions of many other meltdown accident sequences, are given in the Reactor Safety Study (RSS; USNRC, 1975), the Reactor Safety Study Methodology Application Program (RSSMAP) reports (Carlson et al., 1981; Hatch, Cybulskis, and Wooton, 1981; and Kolb et al., 1981), and NUREG-0772 (Denning, 1981b).] The notation used in this appendix and throughout this report to characterize the accident sequences for each of the five reactors (the two RSS reactors and the three RSSMAP reactors) is summarized in Tables A1.1-A1.5.

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*For a hot-leg pipe break close to the reactor pressure vessel, the path through the primary coolant system would be short and the attenuation due to transport through the coolant system often would be negligible; in contrast, for a cold leg pipe break, the path through the coolant system could be relatively long and retention in the coolant system could be substantial for some radionuclides.

**The work reported in this document considered only accidents within the design basis for equipment qualification.

***It should be noted that the descriptions presented here for many of the meltdown accident sequences are dependent upon certain assumptions utilized in the studies on which these descriptions are based.

A.3.1 RSS PWR (Large Containment PWR)*

TMLB'-δ, γ, ε - Loss of reactor coolant system heat removal and loss of all AC power given a transient event; containment failure due to hydrogen burning, overpressurization or basemat melt-through. Reactors are designed so that if offsite power is lost and the diesel generators which provide an emergency source of AC power fail to operate, decay heat can still be removed from the reactor coolant system through the steam generators fed by steam-turbine driven pumps.** With this mode of heat removal failed, however, emergency core cooling pumps, which are driven by AC power, would not operate and the inventory of reactor coolant water would eventually be boiled away through pressure relief valves. Similarly the AC powered containment safety features, such as the containment heat removal system, would not operate. The likelihood of early containment failure by overpressurization in this sequence would be very high and the consequences potentially severe. However, it water were present in either the pressurizer or the pressurizer quench tank, significant fractions of many of the radionuclides released prior to reactor vessel melt-through might be partially scrubbed by that water and prevented from entering the containment atmosphere and thus from escaping to the environment.

V - Interfacing systems loss of coolant accident. Check valves provide a barrier between the low pressure emergency core cooling system and the high pressure piping of the reactor coolant system. In the event that these valves should fail, pressures beyond the design capability of the low pressure system could be imposed on it. The subsequent failure of the system would result not only in loss of reactor

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For example, the timing and the mode of containment failure are both highly dependent upon the assumed containment failure pressure. Different assumptions than those utilized in these studies would have resulted in some not so trivial differences in the descriptions of some of the accident sequences.

*The descriptions in this subsection were adapted from NUREG-0772 (Denning, 1981a).

**It should be noted that not all reactors are equipped with such steam-driven pumps.

coolant, but potentially also in failure of the emergency core cooling system. Since the low pressure piping is located in the auxiliary building, the failure of the reactor coolant system would be external to the containment building. Thus, radioactive material released into the primary coolant system would bypass the containment and thus the containment safety features. In the Reactor Safety Study, this sequence was assessed to be the highest contributor to risk to the human population for the specific PWR plant design analyzed.

S_2C-δ – Failure of containment spray injection given a small pipe break; containment failure due to overpressurization. In the RSS PWR, the early activation of the containment spray recirculation system after a small pipe break would lead to that spray system's failure if too little water were available in the containment sump. Failure of the containment spray recirculation in this plant design would also mean failure of containment heat removal. Therefore, in this accident sequence, the containment building would overpressurize and fail prior to core meltdown. Fuel melting would be delayed approximately one day after the start of the accident. Fission products would be released into a failed containment building. Consequently, the radioactivity released to the environment could be substantial.

S_2D-ε – Failure of the emergency core cooling system given a small pipe break; containment failure due to basemat melt-through. A small pipe break accident would result in a slower depressurization of the reactor coolant system than a large pipe break accident and, in the event of failure of the emergency core cooling system, a more delayed uncovering of the core. Containment safety features would be expected to be operational for this type of accident sequence. Inasmuch as the containment would probably not fail prior to basemat melt-through, the total releases of radioactivity from the containment would be relatively small.

A.3.2 RSS BWR*

TC-γ,γ' – Failure of reactor shutdown systems given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. If the control rods failed to insert and the backup liquid neutron absorber system failed to operate in a transient event requiring reactor shutdown, the reactor power would level off at a heat generation rate well above decay heat. At the estimated power level, the high pressure coolant injection system would not have adequate capacity to match the boiloff of water from the coolant system. The core would eventually become uncovered, heat up, and possibly melt. The large quantity of heat transferred to the suppression pool would result in boiling in the pool, thus preventing further steam suppression and reducing the capability for scrubbing radioactive material. The containment would typically fail by overpressurization before meltdown. The resultant radioactive releases to the environment could be substantial.

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*The descriptions in this subsection were adapted from NUREG-0772 (Denning, 1981a).

TW-γ,γ' – Failure of decay heat removal given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. If the decay heat removal system failed, the suppression pool would be predicted to heat up, boil, and, after an extended period of time, fail the containment by overpressurization. For the specific plant design analyzed, during depressurization of the containment, the emergency coolant pumps would be expected to cavitate with potential to stop delivering cooling water to the reactor vessel. Subsequently, the fuel could become uncovered, heat up, and melt. Fission products released to the reactor vessel would pass through the boiling suppression pool and might not be effectively scrubbed. Fission products released from the fuel to the drywell might pass through the boiling suppression pool or might bypass the pool, depending on the location of containment failure. The consequences of this sequence could be large.

TQUV-γ,γ' – Failure of all makeup water given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. In this sequence, none of the potential sources of makeup water would be available following a transient-initiated shutdown of the reactor. In that situation, steam would be released from the reactor coolant system to the suppression pool through pressure relief lines. The fuel would become uncovered, heat up, and melt. The suppression pool would be subcooled and so could effectively scrub many of the radionuclides released prior to containment failure. The containment would typically fail some time after meltdown. As a result of both more effective scrubbing in the suppression pool and later containment failure, the radioactive releases to the environment would probably be smaller for this sequence than for the other two accident sequences considered for this plant.

A.3.3 Ice Condenser PWR*

S_2HF-γ** – Failure of the recirculation modes of the emergency core cooling system and the containment spray system given a small pipe break; containment failure due to hydrogen burning. For an ice condenser plant, a common cause failure of the two recirculation systems is relatively likely. In particular, the return lines from the upper deck to the containment sump were left closed or became blocked, water sprayed into the upper compartment would remain there. After the ice was all melted and the sump ran out of water, both the recirculation systems would fail. As a result, fuel would become uncovered, heat up and melt. The containment would be expected to fail about the time of reactor pressure vessel melt-through. Neither the ice nor the containment spray system would be available at this point to condense steam or remove radioactive material from the containment atmosphere. Therefore, the consequences to the human population could be substantial.

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*Most of the descriptions in this subsection are based on Carlson et al. (1980).

**This description is based on Denning (1981a).

V – Interfacing systems loss of coolant accident. The discussion given in Subsection A.3.1 about the V sequence for the RSS PWR is appropriate.

S_1H-γ and S_2H-γ – Failure of the emergency core cooling recirculation system given a small pipe break; containment failure due to hydrogen burning. Although the emergency core cooling system would function initially, its failure in the recirculation mode would eventually result in uncovering of the core. Due to the small volume and low design pressure of the ice condenser containment, failure of that containment could very likely occur early in the accident as the result of substantial hydrogen burning. Because of the potential for early containment failure, the releases of radioactivity to the environment could be large, although the early functioning of both the containment sprays and the ice beds would partially mitigate the releases.

S_1HF-γ,δ – Failure of the recirculation modes of the emergency core cooling system and the containment spray system given a small pipe break; containment failure due to hydrogen burning or overpressurization. This sequence would be similar to the S_2HF sequence for this reactor.

TML-γ – Failure of reactor coolant system heat removal given a transient event; containment failure due to hydrogen burning. This sequence would involve loss of both the normal and the emergency means of adding water to the steam generators. Because of this, the steam generators would eventually boil dry and there would be no means of removing decay heat. The resultant increase in reactor coolant system pressure would lead to relief of steam through the safety and relief valves. Water would boil off from the core, leading to core melt. Containment failure would probably occur about the time of reactor vessel melt-through if hydrogen burning occurred. The associated radionuclide releases could be large. However, the operation of the sprays at least initially and the existence of unmelted ice throughout the accident would somewhat attenuate the releases to the environment. In addition, if water were present in the pressurizer quench tank, many of the radionuclides released prior to vessel melt-through might be scrubbed partially by that water.

S_1D-γ and S_2D-γ – Failure of the emergency core cooling injection system given a small pipe break; containment failure due to hydrogen burning. This sequence would be analogous to sequence S_2D for the RSS PWR. The major difference would be that due to the smaller volume and lower design pressure of the ice condenser containment, failure of the containment would probably occur much earlier than in the RSS PWR and would occur as the result of hydrogen burning. (Typically, in the RSS PWR, late failure would occur as the result of basemat melt-through.) Thus, the releases of radioactivity to the environment would be much larger than for the analogous sequences in the RSS PWR.

A.3.4 Mark III BWR*

TPQI-δ - Failure of decay heat removal given a transient event accompanied by a stuck open safety valve; containment failure due to overpressurization. This sequence would be characterized by containment failure preceding melting of the core. Following containment failure and subsequent core melting, the suppression pool would become saturated and the potential for radionuclide scrubbing could be diminished. Thus the radionuclide releases to the environment could be substantial.

TQW-δ - Failure of decay heat removal given a transient event; containment failure due to overpressurization. This sequence would be analogous to sequence TW for the RSS BWR. Containment failure would probably occur before melting of the core. As in the preceding case, the resultant radioactive releases could be large.

TC-δ - Failure of the reactor shutdown system given a transient event; containment failure due to overpressurization. This sequence would be similar to sequence TC for the RSS BWR. The containment would typically fail before meltdown started. Thus, the consequences could be substantial.

TQUV-δ,γ - Failure of emergency core cooling given a transient event; containment failure due to overpressurization or hydrogen burning. This sequence would be similar to sequence TQUV for the RSS BWR, except that containment failure by hydrogen burning is thought to be more likely in this plant. Containment failure by hydrogen burning would be expected to occur shortly after melting of the core whereas containment failure by overpressurization (due to generation of noncondensable gases) would not occur until some time after melting. Both modes of failure would be likely. The radionuclide releases would be smaller for this sequence than for the other three accident sequences discussed above for this plant because of both the attentuation of some of the radionuclides in the subcooled suppression pool and the delayed containment failure.

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*These descriptions are based on Hatch, Cybulskis and Wooton (1981).

A.3.5 Alternate Large Containment PWR*

V – Interfacing systems loss of coolant accident. The discussion given in Subsection A.3.1 about the V sequence for the RSS PWR is applicable.

T_2MQ-FH-γ,ε** – Failure of the recirculation modes of both the containment spray and the emergency core cooling systems given a transient event accompanied by a stuck open safety valve; containment failure due to hydrogen burning or basemat melt-through. This sequence would be analogous to sequence S_2HF for an ice condenser PWR. As in the case for an ice condenser plant, failure of both recirculation modes could be caused by a common mode failure. However, in this reactor, the most likely common cause would be operator failure to realign both systems from the injection to the recirculation mode. Because of the larger size and the higher design pressure for this containment, early containment failure by hydrogen burning would not be as likely as in an ice condenser PWR and so delayed containment failure by basemat melt-through would also be probable in this plant design. The consequences for the former type of failure could be substantial whereas those for the latter type would be relatively insignificant.

S_3FH-γ,ε – Failure of the recirculation modes of both the containment spray and the emergency core cooling systems given a small pipe break; containment failure due to hydrogen burning or basemat melt-through. A stuck-open safety valve would be approximately equivalent to a small break LOCA. Therefore, the discussion just given for sequence T_2MQ-FH is applicable.

T_2MQ-H-γ,ε** – Failure of the recirculation mode of the emergency core cooling system given a transient event accompanied by a stuck-open safety valve; containment failure due to hydrogen burning or basemat melt-through. This sequence would be similar to sequence S_2H for an ice condenser PWR. Because of the larger size and higher design pressure for this containment, the conditional probability of early containment failure by hydrogen burning would not be as high and so there would also be a significant likelihood of delayed containment failure by basemat melt-through for this plant design. The radionuclide releases to the environment could be large if the containment failed relatively early.

S_3H-γ,ε – Failure of the recirculation mode of the emergency core cooling system given a small pipe break; containment failure due to hydrogen burning or basemat melt-through. A stuck-open safety valve would be equivalent to a small LOCA. Consequently, the discussion just given for sequence T_2MQ-H is applicable.

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*These descriptions are based on Kolb et al. (1981).

**Transient-induced LOCA's such as this were not analyzed in detail in the RSS.

T_2MLUO-γ,ε – Failure of the feedwater system, the high pressure injection system and the reactor building cooling system given a transient event; containment failure due to hydrogen burning or basemat melt-through. Failure of the three indicated systems could result from a common cause, in particular, the failure of the low pressure service water (required for component cooling for all the systems). Basically, this sequence would be similar to sequence TML already described for the ice condenser PWR. Because of the larger size and the higher design pressure of this containment, the conditional probability of early containment breaching by hydrogen burning would not be as high and so there would also be a significant likelihood of later containment failure by basemat melt-through in this plant design. The releases of radioactivity in cases of early failure could be substantial. However, if water were present in the pressurizer quench tank, a significant fraction of the radioactivity released before reactor vessel melt-through could be scrubbed from the radionuclide stream to the containment.

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APPENDIX A1

NOTATION

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Table A1.1. Symbols used to characterize RSS PWR accident sequences[a]

A - Intermediate to large loss-of-coolant accident (LOCA).

B - Failure of electric power to engineered safety features.

B′ - Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.

C - Failure of the containment spray injection system.

D - Failure of the emergency core cooling injection system.

F - Failure of the containment spray recirculation system.

G - Failure of the containment heat removal system.

H - Failure of the emergency core cooling recirculation system.

K - Failure of the reactor protection system.

L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.

M - Failure of the secondary system steam relief valves and the power conversion system.

Q - Failure of the primary system safety relief valves to reclose after opening.

R - Massive rupture of the reactor vessel.

S_1 - Small LOCA with an equivalent diameter of about 2 to 6 inches.

S_2 - Small LOCA with an equivalent diameter of about 1/2 to 2 inches.

T - Transient event.

V - Low pressure injection system check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment failure due to basemat melt-through.

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[a] Based on RSS.

Table A1.2. Symbols used to characterize RSS BWR accident sequences[a]

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 inches.

B - Failure of electric power to engineered safety features.

C - Failure of the reactor protection system.

D - Failure of vapor suppression.

E - Failure of emergency core cooling injection.

F - Failure of emergency core cooling functionability.

G - Failure of containment isolation to limit leakage to less than 100 volume percent per day.

H - Failure of core spray recirculation system.

I - Failure of low pressure recirculation system.

J - Failure of high pressure service water system.

M - Failure of safety/relief valves to open.

P - Failure of safety/relief valves to reclose after opening.

Q - Failure of normal feedwater system to provide core make-up water.

S_1 - Small pipe break with an equivalent diameter of about 2 inches to 6 inches.

S_2 - Small pipe break with an equivalent diameter of about 1/2 inches to 2 inches.

T - Transient event.

U - Failure of high pressure coolant injection system or reactor core isolation cooling system to provide core make-up water.

V - Failure of low pressure emergency core cooling system to provide core make-up water.

W - Failure to remove residual core heat.

α - Containment failure due to steam explosion in vessel.

β - Containment failure due to steam explosion in containment.

γ - Containment failure due to overpressure - release through reactor building.

γ′ - Containment failure due to overpressure - release direct to atmosphere.

δ - Containment isolation failure in drywell.

ε - Containment isolation failure in wetwell.

ζ - Containment leakage greater than 2400 volume percent per day.

η - Reactor building isolation failure.

θ - Standby gas treatment system failure.

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[a] Based on RSS.

Table A1.3. Symbols used to characterize ice condenser PWR cccident sequences[a]

A - Large LOCA (>6 in.).

B - Failure of electric power system.

C - Failure of containment spray injection system.

D - Failure of emergency core cooling injection system.

E - Failure of emergency cooling functionability.

F - Failure of containment spray recirculation system.

G - Failure of containment heat removal system.

H - Failure of emergency core cooling recirculation system.

K - Failure of reactor protection system.

L - Failure of auxiliary feedwater system.

M - Failure of power conversion system.

P - Failure of safety/relief valves to open.

Q - Failure of primary system safety/relief valves to reclose after opening.

S_1 - Small LOCA (2 in. $\leqq$ D $\leqq$ 6 in.).

S_2 - Small LOCA (<2 in.).

T - Transient event.

U - Failure of chemical and volume control system.

V - Interfacing systems LOCA.

X - Failure of air return fan system.

Z - Failure of ice condenser system.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment leakage.

θ - Core debris fragmentation upon contact with water in the reactor cavity.

γ - Containment rupture due to hydrogen burning.

δ - Containment rupture by overpressurization.

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[a] Based on Carlson et al. (1981).

Table A1.4. Symbols used to characterize Mark III BWR accident sequences[a]

A - Large LOCA.

C - Failure to render the reactor subcritical.

D - Failure of vapor suppression system.

E - Failure of the emergency core cooling system.

I - Failure of residual heat removal systems after a LOCA (including transient-induced LOCAs).

M - Failure of safety/relief valves to open.

P - Failure of a safety/relief valve to reseat.

Q - Failure of the power conversion system.

S - Small LOCA (the break area is less than one square foot).

T_1 - A loss of offsite power transient.

T_{23} - Any other transient which requires an emergency reactor shutdown.

U - Failure of the high pressure core spray or reactor core isolation cooling system.

V - Failure of the low pressure emergency core cooling systems to provide core flow.

W - Failure of the residual heat removal systems after a transient.

α - Containment failure due to a steam explosion.

γ - Containment failure due to an overpressure caused by rapid hydrogen burning.

δ - Containment failure due to an overpressure caused by gas generation.

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[a] Based on Hatch, Cybulskis, and Wooton (1981).

Table A1.5. Symbols used to characterize alternate large containment PWR accident sequences[a]

A -	Large LOCA (D > 13.5").
B_3 -	Failure of emergency power system.
C -	Failure of containment spray injection system.
D -	Failure of emergency coolant injection system.
F -	Failure of containment spray recirculation system.
G -	Failure of low pressure recirculation system heat exchange.
H -	Failure of emergency coolant recirculation system.
K -	Failure of reactor protection system.
L -	Failure of emergency feedwater system, recovery of power conversion system and high head auxiliary feedwater system.
M -	Failure of power conversion system (normal operation).
O -	Failure of reactor building cooling system.
O' -	Failure of containment spray injection system.
P_1 -	Failure of demand for safety/relief valves.
P_2 -	Failure of safety/relief valves to open.
Q -	Failure of reclosure of pressurizer safety/relief valves.
S_1 -	Intermediate LOCA (10" < D ≦ 13.5").
S_2 -	Small LOCA (4" < D ≦ 10").
S_3 -	Small-small LOCA (D ≦ 4").
T_1 -	Loss of offsite power transient.
T_2 -	Loss of power conversion system transient caused by other than a loss of offsite power.
T_3 -	Transients with the power conversion system initially available.
U -	Failure of high pressure injection system.
V -	Interfacing systems LOCA.
Y -	Failure of reactor building cooling system during injection phase.
Z -	Failure of reactor building coolant system during recirculation phase.
α -	Containment rupture due to a reactor vessel steam explosion.
β -	Containment leakage.
γ -	Containment rupture due to hydrogen burning.
δ -	Containment rupture by overpressurization.
ε -	Containment failure by basemat melt-through.

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[a] Based on Kolb et al. (1981).

APPENDIX B

RESPONSES TO COMMENTS ON DRAFT REPORT

B.1 INTRODUCTION

A peer review meeting was held jointly by the U. S. Nuclear Regulatory Commission and Oak Ridge National Laboratory on March 18, 1982, in Washington, D. C., concerning the work presented in this document. Prior to the meeting, reviewers were provided with copies of both this document and the other report for this project (Niemczyk and McDowell-Boyer, 1982). Comments made on either of the reports during this meeting are summarized here, along with authors' responses to each comment. Due to rigid time constraints, a more in-depth response to each comment was not possible, although some comments certainly warranted more attention. It also was not practical to address pertinent comments within the text of the reports, given the associated time constraints. Comments received after the date of the review meeting are not included in the summary of comments here, and thus, have not been addressed.

The peer review comments received on March 18 were summarized according to the following subject categories. First, general comments are compiled in Section B.2, and addressed, followed by comments regarding accident sequence selection (Section B.3). Comments pertinent to fission product release from fuel are addressed next, in Section B.4, with those pertinent to transport through the primary system and through the containment in Sections B.5 and B.6, respectively.

B.2 GENERAL COMMENTS

Comment: The review was rushed.

Response: The entire project was performed on a very short time scale and all aspects of the project, including the review, were rushed.

Comment: The purpose of the study was not clear.

Response: The purpose of the study, as given by the funding agency, is stated in Section 1.1 of Chapter 1 of this report.

Comment: Several people commented on the lack of a start-to-finish description for the estimation of source terms for any accident.

Response: The details of almost all the procedures used for the illustrative estimation of source terms, along with the intermediate results of those procedures, are presented in Appendices B-D of the other report for this project, ORNL/TM-8275 (Niemczyk and McDowell-Boyer, 1982). However, they are not organized on a start-to-finish basis, as this was not perceived as a priority in completing the work under the given rigid time constraints. The overall set of procedures, as well as the sources of all the input data, are summarized in Chapter 3 of the other report.

Comment: There were excessive conservatisms in all steps of the illustrative calculations.

Response: Like truth and beauty, what is "excessively conservative" is partly in the eye of the beholder. The goal was to estimate, in as realistic a manner as possible, the source terms which might be anticipated for various types of accidents. Unfortunately, there are large uncertainties associated with many aspects of source term estimation. When what was realistic was not obvious, the tendency, in performing the estimations, was to choose values lying on the high sides of the indicated ranges of values. In no instances were safety margins used or worst-case type analyses performed for individual accidents.

Comment: A lot of source term work is going on and there will be break-throughs in the near future which will substantially improve the quality of source term estimations.

Response: A lot of work is going on which ultimately should improve at least certain parts of the art of source term estimation. However, there are several areas of large uncertainties which must be "solved" before the source terms for all accidents can be substantially improved. Such problem areas include the estimation of thermal-hydraulic conditions, the consideration of radionuclide and aerosol release rates, the description of transport through the primary coolant system, and the estimation of transport within the containment. Unless there are at least a few important break-throughs very soon, and this does not seem likely, there will continue to be large uncertainties associated with most source term estimates.

Comment: There are a large number of problems associated with the use of the MARCH code. These may be important because MARCH is used for many aspects of source term estimation.

Response: The difficulties with MARCH are well known (Rivard, 1981). However, there are not currently any other codes which can be used instead of MARCH for all types of accidents.

Comment: The discussion of the limitations, gaps in knowledge, and uncertainties should be expanded and their potential impacts noted.

Response: The authors agree completely. Unfortunately, the extremely limited scale of this project restricted the amount of discussion which could be devoted to these very important topics. Some discussions of these topics are presented in Appendices B-E of the other report for this project, ORNL/TM-8275. Much more effort needs to be expended in this area in general.

Comment: Synergistic effects could increase or decrease the effects of many transport processes. They should be discussed further.

Response: The authors agree. Unfortunately, certain of these effects have been understudied in the past and it was far beyond the scope of this study to consider such effects in more detail than had been done previously.

Comment: In the absence of good technology, rapid generalizations based on slender evidence should be avoided.

Response: The authors agree. However, it should be pointed out that regulations concerning very rare events, such as meltdown accidents at nuclear plants, cannot always be based on as thorough an information base as is desirable. In such cases, it makes no sense to ignore the potential hazard just because all the relevant information is not available. In these cases, the available information must be used as skillfully as is possible. Rapid generalizations should always be avoided.

Comment: The Zion/Indian Point studies should be looked at for their handling of uncertainties.

Response: The Indian Point study was not available during this project. The Zion study, which became available during this project, contained no new information about the handling of uncertainties.

Comment: Radioactive decay should be considered in source term estimations.

Response: The authors agree. However, this has not been done in the past. The very limited scale of this project precluded any extensive consideration of the problem here.

Comment: The chemistry of all species should be considered in source term estimations.

Response: The authors agree. However, this has not been done in the past for most species. The very limited scale of this project precluded any detailed consideration of the problem here.

Comment: All species except iodine, and perhaps cesium, should be ignored in source term calculations.

Response: The goal of the project reported in this document was to develop assumptions appropriate for realistically estimating source terms. Elimination of most radionuclides from consideration represents a considerable step backward from such an approach. In addition, the dose to the equipment inside the reactor building is dominated by different radionuclides than the dose to the human population outside the plant. Thus, what are adequate surrogates for all the radionuclides for the human population are not necessarily adequate surrogates for the equipment.

Comment: The differences of opinion on the chemical problems should be clarified.

Response: There are many areas for which there does not seem to be adequate chemical information to make reliable predictions. Known differences of opinion are discussed in Appendices B-D of the companion report, ORNL/TM-8275 (Niemczyk and McDowell-Boyer, 1981). Although the authors attempted to at least mention conflicting viewpoints when they were aware of them, the very limited scale of this project necessarily restricted the number of such issues which could be addressed and the depths in which any of them could be treated.

Comment: Less reliance should be placed on unpublished information.

Response: Very little reliance was placed on publicly unavailable information except where it was necessary to do so to incorporate very recent and not yet published work in order to appropriately address the overall problem. In fast-moving fields, such as certain aspects of source term estimation, much of the most recent information at any given time has not been published in the journals but instead has been published only in laboratory reports, if at all. The prejudice of some against using information from such publicly available laboratory reports serves only to condemn source term work to needless conservatism.

Comment: The documentation should include provision of definitive settings for Emergency Action Levels.

Response: This is outside the scope of the project described in this document.

Comment: All relevant research work should be discussed.

Response: It is outside the scope of the project reported in this document to describe in detail all of the work related to source term estimation which is currently underway. A number of projects which are ongoing are referred to in Appendices B-D of the other report associated with this project, ORNL/TM-8275.

Comment: The work should be continued. A standing peer review committee should be set up to guide the writing of the final report.

Response: The project described in this document and in ORNL/TM-8275 was a short-term project intended to provide interim guidance while the results of a separate, much more extensive project at Battelle Columbus Laboratories was completed. A permanent standing committee related to all source term considerations might help future source term work by helping to disseminate the most recent findings to all interested parties.

B.3 ACCIDENT SEQUENCE SELECTION

Comment: What was the basis for the selection of the accident sequences considered for the study?

Response: The ranges of sequences considered for emergency planning and for equipment planning were defined by the funding agency. The sequences singled out for explicit consideration for emergency planning were chosen by the authors from sequences formerly described and investigated in detail by others. The specific sequences investigated for emergency planning were chosen because of the availability of previously performed calculations which could be used as the basis for obtaining updated source term estimates. The sequences considered for equipment qualification were chosen because of the constraint placed upon this project that only worst-case sequences within the design basis be investigated for that purpose.

Comment: Event V should not be considered since post-TMI modifications have reduced the probability of this accident.

Response: This project was not concerned with estimating the probabilities of accidents, but rather was concerned with estimating the magnitudes of potential releases. However, it might be noted that the probability of the V event is highly dependent on both the plant design and the procedures followed and varies greatly among plants. It cannot currently be said that the V event has a relatively low probability at all or even in most nuclear plants.

Comment: Steam explosions should not be considered.

Response: As is noted in the report, steam explosions were not considered.

B.4 FISSION PRODUCT RELEASE FROM FUEL

Comment: Do the MARCH code calculations of temperature histories of the core during accident sequences consider both radial and axial heat transfer by radiation?

Response: An evaluation of MARCH, in terms of either the validity or uncertainty associated with this code, was much beyond the scope of the present study. Rather, MARCH output was used for the illustrative calculations because it was available at the time of the present study and because the code is documented. However, the documentation of MARCH (Wooton and Avci, 1980)indicates that such heat transfer is not considered in the code.

Comment: After core slumping, the surface-to-volume ratio of the core is altered dramatically from that associated with fuel rods and that associated with a molten pool of core material. Is the NUREG-0772 method of estimating fission product release from fuel applicable to releases after slumping?

Response: The limitations in application of the NUREG-0772 method for estimating release from fuel to post-slumping conditions before vessel failure are not explicitly discussed in that document (Wichner et al., 1981). It is questionable whether the experimental data from which release rate coefficient curves were generated should be used to estimate release rates during the post-slumping phase of an accident since geometries will vary so dramatically between experimental and full-scale accident conditions. However, in lieu of more pertinent, published experimental data, the release rate coefficients from NUREG-0772 were used to estimate releases during this phase of an accident sequence. A surface-to-volume ratio correction factor was not used to adjust these estimates, although preliminary and as yet unpublished experimental results indicate that such a correction will improve the estimates (Packer, 1982). An illustrative calculation utilizing such a correction factor was made in the document which provides the technical basis for the present document (Niemczyk and McDowell-Boyer, 1982), indicating much lower estimated releases during this post-slumping phase, although total fractional releases change very little except for fractional releases of Sr and Ba, which change by less than a factor of three. It should be noted that other factors affecting release may be altered from the pre-slumping phase after slumping, yet are not addressed by such a correction factor.

Comment: Has the decay heat depletion of the core, due to fission product release from fuel, been considered in temperature history calculations for the core?

Response: Documentation of the MARCH code (Wooton and Avci, 1980) indicates that decay heat depletion is accounted for in temperature calculations, using WASH-1400 (U. S. Nuclear Regulatory Commission, 1975) best-estimate release fraction values, with a somewhat modified time-dependence due to the inclusion of the gap release component with the melt release component. Ideally, release rate fractions as a function of time from the present study should have been used to estimate temperatures which in turn would have been coupled with these latter time-dependent release fractions, but the time-constraints and scope of the present study did not allow generation of new MARCH output for the accident sequences considered.

Comment: Is there a consistency in the treatment of fission product aerosols and structural material aerosols?

Response: Structural materials in the present document are taken to represent stainless steel and alloys present within the core and supporting the core, excluding the absorber material in the control rods and the zircaloy cladding. Structural material aerosols are therefore estimated using the same procedure as used for fission products; i.e., using a release rate fraction as a function of temperature in conjunction with an estimated inventory. Because iron is the predominant structural material specie, a release rate fraction for iron, from the SASCHA data (Albrecht and Wild, 1981) was used to estimate structural material aerosol production.

Comment: The calculation of releases from fuel did not consider the rationale in the Reactor Safety Study (RSS). The NUREG-0772 approach is no more valid than the RSS with respect to Te, Ba, and Sr releases.

Response: The two approaches presented in the RSS and NUREG-0772 documentations are somewhat different, yet rely on the same experimental data with the exception of the NUREG-0772 consideration of the recent SASCHA data (Albrecht and Wild, 1981). In both cases, much uncertainty lies in the release estimates generated, as a result of the necessity of extrapolating from bench-scale experimental conditions to full-scale in-reactor conditions. What should be noted, however, is that the results generated for each fission product category by both methods are quite similar, and certainly lie within the bounds of uncertainty associated with the estimated values. Furthermore, it is important to update estimates when more experimental data becomes available, which was accomplished to some extent in NUREG-0772. Fortunately, experimental work is ongoing in this field with which present estimates and estimation procedures may be improved.

Comment: The German SASCHA data is not suitable for estimating absolute release rates, but only for estimating relative release rates.

Response: Limitations of experimental data used to estimate release fractions have been noted in the document which discusses the technical basis for the present release estimates (Niemczyk and McDowell-Boyer, 1982). The SASCHA data has been questioned as to its adequacy due to the manner in which fission product simulants are incorporated into the fuel pellets, and due to the manner in which the heat-up experiments were conducted. However, limitations, possibly equally as serious, have been noted for other experimental data, particularly with respect to the small scale on which they were conducted and in some cases, the absence of cladding and use of other than fuel rod geometries.

Comment: Was the solubility of lanthanum considered in the estimated release from fuel during leaching?

Response: The empirically-derived leaching release fraction of uranium was assumed to approximate that of lanthanum in lieu of more specific data. This was assumed for the sake of consistency with fission product categories used in the RSS and adopted in the present study.

B.5 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

Comment: Several people wanted to know the details of the primary coolant system retention estimations.

Response: The very short time scale of the project reported here precluded a detailed write-up of the method used for estimation of primary coolant system retention. All of the basic assumptions used in that method are discussed in Appendix C of the other report for this project, ORNL/TM-8275.

Comment: The retention of aerosols in the primary coolant system was overemphasized. The retention of materials on internal coolant system surfaces was underemphasized.

Response: Retention of radionuclides in the primary coolant system has been understudied in the past. In fact, typically, it has not been treated in a quantitative manner at all. Recently, however, there have been some calculations performed which indicate that for certain accidents, aerosol and/or vapor processes might result in substantial retention of some materials in the primary coolant system (Gieseke and Kuhlman, 1981). Although the initial investigations concentrated on retention of aerosols, further work is currently being conducted on retention due to various vapor-related processes, including chemisorption onto coolant system surfaces. The study of retention in the primary coolant system is an area currently experiencing substantial theoretical and experimental activity and probably will be for some time to come. At this point, both aerosol and vapor processes need to be considered much more extensively than they have been in the past.

Comment: The chemistry of tellurium in the primary coolant system should be considered.

Response: The interactions of tellurium in the primary coolant system are currently being investigated both theoretically at Battelle Columbus Laboratories and experimentally at Sandia National Laboratories. The results of that work are just becoming available. Unfortunately, none of that work was available at the time this project was being performed. Little other applicable work has been done. The chemistry of tellurium, like the chemistries of many other elements, deserves much more attention than it has received in the past.

Comment: The aerosol residence times in the primary coolant system are of no importance because aerosol concentrations would always be so high.

Response: Both limited-core-damage accidents and meltdown accidents were considered in the project described in this document. Thus, for at least some of the accidents, the aerosol concentrations in the primary coolant system would be relatively low. And even for meltdown accidents, the aerosol generation rates are highly uncertain. Consequently, even within the reasonable ranges of possible aerosol concentrations for meltdown accidents are some relatively low concentrations.

Comment: Resuspended aerosols would be too heavy to be transported.

Response: Transport of resuspended aerosols would probably not be a major means of movement. However, the possibility of such transport of some materials cannot be excluded.

Comment: Reactions of iodine in the primary coolant system were not considered.

Response: The formation of CsI was considered. Other reactions involving CsI were not explicitly considered but the possibility of conversion of iodine from CsI to elemental form was included implicitly. More work needs to be done in this area.

Comment: The estimates of aerosol retention in the primary coolant system in NUREG-0772 are underestimates for many reasons and not overestimates as suggested in this report.

Response: For the many reasons indicated in Appendix C of the other report for this project, aerosol retention might be much lower than the retention estimates implied in NUREG-0772 (USNRC, 1981). The primary reason aerosol retention might be higher than indicated in NUREG-0772 would be that the aerosol input rates for the calculations in that study might be too low. That the input rates used in NUREG-0772 were too low, however, is not at all obvious. (It should be noted that the aerosol generation rates are among the most important but least well known factors in source term estimation.)

Comment: Steam flow rates through the primary coolant system are always low except prior to fission product evolution and after slumping. During the critical fission product release period, the steam flow rates always would be low.

Response: The reader is referred to Appendix C in the other report for this project, ORNL-8275, which addresses this topic.

Comment: Delayed operation of the emergency core cooling system would quench the core and attenuate the aerosols.

Response: The amount of attenuation would be highly accident and plant dependent. Most accidents probably would not involve delayed operation of the emergency core cooling system.

Comment: Scrubbing of aerosols and vapors in the primary coolant system is inadequately treated in this report.

Response: In general, scrubbing of aerosols and vapors in the primary coolant system and in the containment have not been treated adequately either theoretically or experimentally. Some work in this area is currently in progress at Battelle Columbus Laboratories.

Comment: The numerical ranges for retention in the primary coolant system are not reasonable.

Response: The ranges reflect in part the large uncertainties associated with estimations of retention in the PCS. They also reflect plant to plant variations.

Comment: Resolving the uncertainties would tend to reduce the escape fractions.

Response: It would tend to reduce the escape fractions for some species in certain accident sequences.

B.6 TRANSPORT WITHIN THE CONTAINMENT

Comment: The CORRAL code does not represent the state-of-the-art.

Response: It is true that CORRAL does not represent the state-of-the-art. However, it does form a consistent basis for modeling the post-accident transport of radionuclides in the containments of LWR's. Currently, other codes can only be used to perform parts of calculations and not to perform entire calculations. As the other, more advanced containment codes currently under development are completed, they may significantly improve the state-of-the-art.

Comment: The effect of the presence of large quantities of water in the containment was not properly modeled.

Response: None of the computer codes used for considering behavior in the containment includes the effects of the presence of large quantities of water. In general, this effect has been understudied.

Comment: The steam environment in the containment should be modeled better.

Response: Currently, only one computer code, NAUA, treats arbitrary steam environments in the containment. Sufficient data is not yet available to use that code in a predictive manner. As was noted in the other report for this project, the treatment of steam environments, especially treatment of condensing steam environments, needs more attention than it has received in the past.

Comment: There is not enough analysis of the containment failure modes to pin down the time when radionuclides would be released to the atmosphere. The effects of scaling in the containment failure analysis should be considered.

Response: There was no analysis of the timing of containment failure in this project. Such an analysis was beyond the scope of this project. As was noted in the other report for this project, ORNL/TM-8275, the timing of containment failure is one of the most important but yet least well known factors affecting the releases to the environment. Currently, in the estimation of source terms, containment failure is assumed to occur at a certain time in the accident scenario and the analysis is performed using that assumption. The assumptions used concerning containment failure are based heavily on engineering judgments. This study utilized the assumptions of others in the area of containment failure analysis.

Comment: Many problems are associated with the use of CORRAL.

Response: A number of these are mentioned in Appendix D of the other report for this project, ORNL/TM-8275.

Comment: Clogging of containment leak paths was not considered.

Response: Clogging of leak paths from the containment was discussed in Appendix D of the other report for this project, ORNL/TM-8275. Clogging was not treated quantitatively for the reasons stated in Appendix D.

Comment: The NAUA results appear to be misplotted.

Response: This comment refers to Figure D.1 in Appendix D of the other report for this project. As is noted in Appendix D, this figure was taken from the Technical Bases Report (USNRC, 1975). The authors of this report have been unable to obtain further information from the commenter about his objection to the indicated figure.

Comment: The report says that LMFBR codes are not applicable to LWR accidents.

Response: The other report for this project, ORNL/TM-8275, states that great care must be exercised in using codes developed for treating sodium oxide aerosols in LMFBR's to consider aerosols in condensing steam environments in LWR's. The dissimilarities of the two environments are numerous and cannot be disregarded. In addition, the LMFBR codes do not include consideration of the effects of any of the LWR-engineered safety features.

Comment: Adjusting CORRAL results to account for acknowledged problems is technically indefensible.

Response: There is currently no single computer code which can be used for all considerations related to transport of radionuclides in the containment. To obtain estimates of transport in the containment, it is adviseable to use several codes to gain various insights into the overall problem. To take the results of a code such as CORRAL, which has acknowledged problems, and not to take those problems into account either by using other codes or by applying corrections to the results seems indefensible.

Comment: It is stated that "because of the somewhat arbitrary nature of the definition and choice of the accident sequences ... problems with many of the detailed models ... are not overwhelmingly important." This statement is not defensible.

Response: The questioned statement, which is in the other report for this project, actually reads "... are not necessarily overwhelmingly important for the overall description of any accident." Most people are unaware that parameters used in accident descriptions are not estimated using detailed models but instead are estimated somewhat arbitrarily using engineering judgment. Thus, to dwell only on the problems in the models and related codes ignores other fundamental problems which are very important to source term estimation.

References for Appendix B

Albrecht, H., and H. Wild, 1981. "Investigation of Fission Product Release by Annealing and Melting of LWR Fuel Pins in Air and Steam," paper presented at the *Topical Meeting on Reactor Safety Aspects of Fuel Behavior*, August 2-6, 1981, Sun Valley, Idaho.

Gieseke, J. A., and M. R. Kuhlman, 1981. "Fission Product Transport in Primary System to Containment," Chapter 6 in *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, NUREG-0772, USNRC, Washington, D. C.

Niemczyk, S. J., and L. M. McDowell-Boyer, 1982. *Technical Considerations Related to Interim Source Term Assumptions for Emergency Planning and Equipment Qualification*, ORNL/TM-8275, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Parker, G. W, 1982. Private communication, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Rivard, J. B., 1981. *Interim Technical Assessment of the MARCH Code*, NUREG/CR-2285, SAND81-1672, Sandia National Laboratories, Albuquerque, New Mexico.

U. S. Nuclear Regulatory Commission, 1975. *Reactor Safety Study*, WASH-1400 (NUREG-75/014), USNRC, Washington, D. C.

U. S. Nuclear Regulatory Commission, 1981. *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, NUREG-0772, USNRC, Washington, D. C.

Wichner, R. P., T. S. Kress, and R. A. Lorenz, 1981. Chapter 4 in *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, NUREG-0772, USNRC, Washington, D. C.

Wooton, R. O., and H. I. Avci, 1980. *MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual*, NUREG/CR-1711 (BMI-2064) Battelle Columbus Laboratories, Columbus, Ohio.

APPENDIX C

PEER REVIEW COMMENTS

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 2 1982

MEMORANDUM FOR: L. G. Hulman, Chief, Accident Evaluation Branch, DSI

THRU: W. F. Pasedag, Leader, RAS, AEB, DSI

FROM: Mohan Thadani, AEB, DSI

SUBJECT: PEER GROUP MEETING TO DISCUSS ORNL DRAFT REPORT ON "INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATIONS" (FIN #B0767)

The subject meeting was held in conference room P-118 on March 18, 1982, to review the draft report of the ORNL contract work (FIN #B0767) sponsored by the Accident Evaluation Branch. Enclosure 1 is a list of the attendees. The meeting was organized to allow ORNL to present a summary of the work performed and reported in the draft report. Enclosure 2* is a copy of the viewgraphs presented at the meeting. The ORNL presentation was followed by presentation of prepared comments by the peer group attendees. Enclosure 3 is a copy of prepared peer comments. Detailed notes of the meeting were taken by Sue Niemczyk, L. McDowell-Boyer, Walt Pasedag and myself. All the verbal and prepared comments were culled from these notes and have been organized into different groups according to the aspect of the release involved in the accident. Each comment was further rated according to the time required by ORNL to respond to it in the final report. The categorized comments are presented in Enclosure 4.

The meeting started promptly at 9:00 a.m. and ended at 3:40 p.m. A large number of good suggestions were made. Careful consideration should be given to the significant comments affecting any future source term decisions.

Mohan Thadani

Mohan Thadani
Radiological Analysis Section (B)
Division of Systems Integration

Enclosure:
1. List of Attendees
2. Viewgraphs
3. Prepared Peer Comments
4. Categorized Comments

*Enclosure 2 is not included in this Appendix

cc:
HDenton
RMattson
RWHouston
SHanauer
RVollmer
DEisenhut
ABice
GKelly
TSpeis
WButler
MTaylor
JLong
ASpano
DRoss
SBassett
RBernero
GArlotto
MSilberberg
RSherry
RBlond
MCunningham
BGrimes
EWilliams
SNienczyk - ORNL

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ENCLOSURE 1

NUREG-CR-2629 PEER REVIEW MEETING

NAME	ORGANIZATION	TELEPHONE
Richard C. Vogel	EPRI	415-855-8910
Mohan Thadani	NRC	301-492-8941
Walter Pasedag	NRC	301-492-4993
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A. L. Lotts	ORNL	FTS-624-0422
Edward Warman	Stone & Webster	617-973-6510
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Rick Sherry	NRC/RES	FTS-427-4266
Mel Silberberg	NRC/RES	FTS-427-4266
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Wayne Houston	NRR/DSI	301-492-8026
Jim Cunnane	BCL	614-424-4082
Tom Kress	ORNL	615-574-0558 FTS-624-0558

NAME	ORGANIZATION	TELEPHONE
Michael Kuhlman	BCL	614-424-5393
Jacques Read	NRC/AEB	301-492-8301
Sudarshan K. Loyalka	UMC	314-882-8201
L. S. Gifford (Representing W. C. Sitirley)	GE	301-654-0011 408-925-1192
Jerry Hulman	NRC	301-492-7880
P. S. Rohwer	ORNL	FTS-626-2100
R. O. Chester	ORNL	FTS-626-2103
L. M. McDowell-Boyer	ORNL	FTS-626-2102
S. J. Niemczyk	ORNL	FTS-626-2116

ENCLOSURE 2

Not included in the Report.

119 [redacted]

ENCLOSURE 3

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2 Acoma Lane
Los Alamos, NM 87544

672-37

March 26, 1982

Mr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration, NRR
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Hulman:

I enclose herein my comments on the draft report, "Interim Source Term Assumptions for Emergency Planning and Equipment Qualifications." A large fraction of my primary recommendations on the above draft report apply equally to the revision or update of NUREG-0772, "Technical Bases for Estimating Fission Products Behavior During LWR Accidents." The applicability of several points will be obvious in the text.

My primary and most important comment derives from the fact that the study is highly "conservative" in nature instead of being realistic or "best estimate" in nature. The matter of emergency planning is especially significant because the actions and possible movements of the public are involved. For these reasons it is very important that most careful estimates be obtained for the NRC to consider before new rules and regulations are created.

My comments derive from a philosophy for safety analyses to which I (and I believe many others) subscribe, to wit: If an analysis of, e.g., a reactor accident sequence of events is performed with a conservative calculation and with conservative assumptions throughout and the result shows no threat to public or plant, a useful product is created. Even though a margin of safety has not been obtaine the result is still useful for regulations and to the reactor owners and operators. However, if the same analyses produces a result that shows a threat (e.g., radiological), to the public or plant, the study is of little value and may indeed, be useless because without additional information the analyst cannot say whether the threat is real or merely derives from and is a function of the calculation and the assumptions (all conservative as stated above). Regulatory actions taken on the basis of the second case may be wrong, and, in fact, may create a new hazard, nonexistent before the action was taken.

On the other hand, if an analysis of the same accident sequence is performed with a realistic or best estimate program with comparably good assumptions and if the analysis shows no threat, the result is useful as before, but now some idea as to the margin of safety is

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obtained and regulatory action can be based on better information. If the realistic study shows a threat to public or plant, the analyst can be much better assured that the threat is real and not an artifact of the calculations. Again, actions, if any, can be based on solid information.

A final point on this subject is worth mentioning; the second analysis is based, presumably, on a best model of the real physical and chemical laws. Given this state of affairs, the necessary programmatic constants and assumptions can be shaded to the conservative or non-conservative limit of known error to find, in this manner, a margin of safety that can be believed.

It appears to me that this discussion is appropriate to the document under review, "Interim Source Term Assumptions for Emergency Qualifications." As was abundantly made clear in the peer review discussions on March 18, 1982, the study, and assumptions found within the analyses, were conservative in nature, some very strongly so. The final results showed a possible radiological threat to the health and safety of the nearby population and, if the data are accepted by the NRC, some actions could be required by the accident, actions which might lead to real hazards. The possibility of such action requirements at some future time also could prove deleterious to the nearby residents, as was made abundantly clear at Three Mile Island in 1979. Thus, because the study is conservative, there is no way to know (from this study) whether the threat predicted is real or simply an artifact of the study; possible actions could be taken that might be unnecessary and even harmful.

My foremost recommendation, therefore is that this study not be accepted as a basis for decision making by the NRC; however, because of the importance of the subject (emergency planning) and because of notable study effort is underway with a staff in place which has produced a document of importance I further recommend that the study be continued with the specific direction from the NRC to produce a realistic estimate of risk (and consequences) for emergency planning purposes. Finally, to assure that the study have input from all concerned parties, I recommend that a steering committee be created of persons from the NRC, DOE, industry (vendors, utilities, architect and engineering firms), National Laboratories, etc. This last suggested action would serve at least two basic purposes: The probability that all relevant experimental and theoretical information be included, would be increased significantly and the chance that the study be realistic would be greatly improved --- at the least, if at some point divergent points of view existed, alternate paths of analysis could be followed.

Such alternate paths, if followed rigorously, could very possibly, predict conditions that could be tested experimentally, thus resolving serious unknowns in our understanding of reactor accidents.

These last points are equally, perhaps even more, relevant to the "update", "revision", or whatever of NUREG-0772 that was mentioned briefly on March 18, 1982. As I recollect from the peer review of this latter document (0772) about a year ago, some criticisms were severe and in the same vein as those mentioned on March 18, 1982. Thus, direction by the NRC for a realistic analyses, creation of a broadly based steering committee, and allowing provisions for alternate analytic paths are matters of extreme importance for this revision of NUREG-0772. I offer these recommendations to the NRC for NUREG-0772 because I believe that the document may prove to be very important for future rules, regulations, and actions taken by the NRC and the industry. To obtain broadly based support for the document during its creation would serve to make it far more effective in producing useful results.

DETAILED COMMENTS:

My detailed comments on this document derive from my personal philosophy and recommendations mentioned above. My comments are scarcely complete (the document is large) and I apologize for the preponderance of criticisms. Really, a major and important piece of work has been accomplished and the authors are to be complemented. I further caution that these comments are in no particular order, and derive more from examination of the appendices than from the body of the document.

1. It is not evident that the several containment systems experiments have been factored into the analysis. These are far too significant to ignore.

2. The words release or escape (of fission products) are often used in a context that does not explain from where to where. For example, escape or release from fuel to the primary system may have been intended, but escape from fuel to the environment could be inferred.

3. The boiling temperatures of CsI, CsOH are substantially higher than for Cs and ~~I2~~ I_2. This point is not made in the discussions, may not have been considered.

4. Discussion of the chemistry of iodine in relation to cesium (CsI) demands an examination of the chemistry of all the important fission products. (E.g., CsOH is the predominant form of cesium that is not tied up with a halogen).

5. I doubt that a full core would ever melt even in a situation with minimal amounts of steam. Black body radiation can remove heat at a very rapid rate. The problem should be examined quantitatively. Steam cooling should be taken into consideration also.

6. The results for strontium and barium are surprising, but I found no discussion of the thermodynamics of Sr, SrO, SrOH, etc.

7. A conclusion of NUREG-0772 was the methyl iodide is much less abundant than previously assumed. This was, apparently ignored.

8. Would consideration of the SL-1 accident assist in the analysis of the postulated V event. What are the conditions in the auxiliary building when and if fission products begun to reach the building? Steam atmosphere, wet walls, wet floors? Because of the extreme solubility of CsI and CsOH, these conditions should be considered.

9. Tables D.1 and D.2 show some assumptions that are incorrect or negative in tone. E.g., "Radioactive decay probably not important;" Containment sprays "removes ~~from~~ some aerosols and some iodine." The experiments are nearly dramatic, they apparently were not considered.

10. The importance of steam, both dry and wet, in the primary system is never mentioned.

11. The CORRAL code was used to describe actions in the containment. Because this code has been so widely criticized some justification for its use is needed, some comparison to experiments or accidents would be helpful. Why not test and use the NAUA program which is said to model more physical effects?

Can the TRAP code predict plateout of various chemical species on cool surfaces and handle the problem of condensing steam and how it "traps" vapors and aerosols? Where has it been tested?

12. Page D-23, Escape from containment mentioned as taken from CORRAL results from WASH-1400. These data must be at least eight years old. Newer analyses, the ZIP study, are said to show the containments to be much stronger than assumed 8 years ago.

13. The RSSMAP study is mentioned, but no results are given. Most of the reviewers have not had access to this work.

14. Page D-25, the chemistry of all important fission products should be discussed. It is not clear where the chemistry of iodine was taken into account.

15. Figure D.2: The notion that kilogram quantities of aerosols could go through the earth filter, below the basemat needs to be justified. In any case is not TMLB'-ε a very small contributor to risk?

16. The statement at the bottom of page D-36 is very interesting. If such rapid steam condensation occurs, the containment pressure would be very low and no failure would occur. Hence, no release of fission products.

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17. The conclusion at the top of page D-37 should be unsatisfactory to the NRC. Experiments have been performed that are applicable.

18. Page D-38, D-39; the conclusion on modifications to apply because of leak size are unsatisfactory and much too qualitative. If this point is important, and I think it is, the discussions must be much more complete.

19. Page D-46; the timing prescriptions from the RSS are mentioned as being used. Because much new information exists, why use this prescription?

20. Page D-47; the meaning of "vaporization" is not clear. Vaporation of what?

21. Page D-47; again CORRAL is accepted without comment or criticism.

22. Page E-1 begins by asserting that "best estimate" for the source term is obtained. I submit that this is not the case. The source term is very conservative.

23. Table E.1; for the PWR, the RSS categories PWR-2 (V only), PWR-2 and PWR-3 dominate overwhelmingly. A detailed discussion of typical events in these categories would lead to a great deal of clarification and make discussion much easier and more profitable.

I have personal doubts about the vaporization of some species usually regarded as refractory. The high temperature thermodynamics must be considered in order to establish the appropriate chemical compounds.

24. Page E-14; some of the limitations of some programs are mentioned, e.g., chemistry, radioactive decay. This should not be an excuse to not consider such factors. It might be better in some circumstances to ignore the codes.

25. Page E-19; in the conclusion the authors suggest that "nonconservative aspects of source term estimation abound in both RSS and in subsequent source term work. If indeed these "abound" they can be tabulated easily. I suggest that a useful contribution would be to create a tabulation of both conservative and nonconservative aspects. To start such a list I suggest the following aspects that I believe are a part of WASH-1400.

a. Fuel at melting temperature at any point in the core is equivalent to full core melting.
b. If any fuel melts all fuel melts.
c. For some accidents there is no attenuation of fission product source term core, vessel, piping, containment, or auxiliary building.

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d. Iodine is assumed to be a near-perfect gas instead of a reactive species, probably combined with cesium to form CsI.
e. The chemistry of other fission products (e.g., cesium) also is ignored.
f. There is no consideration of the effect of condensing steam in the primary system or containment.
g. The mode of containment failure is assumed to be instantaneous and large.

To summarize, I find the analyses to be unacceptably conservative and unrealistic. Nevertheless, I repeat my three recommendations offered at the beginning of this letter. I believe that the matter is very important and should continue to a successful conclusion. The momentum in being should not be lost.

I repeat a suggestion made, I believe, By Dr. Vogel of EPRI: A small group of accident sequences should be discussed in detail, giving all assumptions, calculational codes used with their advantages and disadvantages, results, etc. This by itself would be a big step forward and contribute immensely to future discussions.

I believe that it is abundantly evident that this "Source Term" issue is not going away in the near future. Some workers in the field believe that large changes are at hand, factors of 10, 100, or more in some cases, in addition to the very large factors in TMI-like accidents. I urge that these matters be pursued until fully clarified. The issue is far too important to give less than a full effort.

In response to your direct question to the group on March 18, "Do data exist that can justify a reduction in the source term assumed for emergency planning" (if I paraphrase accurately). My response is yes, data do exist to justify a modification (reduction for most chemical species) in the source term. These factors can be obtained by careful detailed and quantitative consideration of aspects such as aerosol behavior, fuel melting, plateout on cool surfaces, the action wet steam, containment experiments, chemistry of the most important fission products, time and modes of containment failure, etc. I further believe that emergency sources of power (e.g., fire trucks) and emergency sources of water (e.g. city water supplier) for containment sprays should be a part of the analyses of postulated accidents.

I thank you for inviting me to contribute to this "peer" review. I look forward to more interactions in the future.

Sincerely,
W. R. Stratton
William R. Stratton

WRS:dp

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EPRI

March 26, 1982

Mr. L. G. Hulman
Chief
Accident Evaluation Branch
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Hulman:

This letter is being submitted in response to your suggestion made at the peer review meeting (March 18) of ORNL/TM-8274 (NUREG/CR-2639), "Interim Source Term Assumptions for Emergency Planning and Equipment Qualification" and ORNL-TM-8275, "Technical Considerations Related to Interim Source Term Assumptions for Emergency Planning and Equipment Qualification".

It will be recalled that I left written comments prepared by the EPRI staff at the meeting. The purpose of this letter is to offer further material for your consideration.

Only a superficial review of the methods used in ORNL/TM-8274 and 8275 is now possible because the calculations for any specific accident scenario cannot be followed from beginning to end. We therefore feel that ORNL's Task 5, "A Formal Peer Review" of the program cannot be finished until completely visible "beginning to end" calculations for several of the more important accident scenarios are included in the documentation. Specific listings of the numerical input to the codes used as well as a discussion of the under-lying principles and assumptions involved in the codes should be presented. A discussion for the rationale for the selection of the input data is needed. For example, it is noted that the fission product release rate data of NUREG-0772 are used but were "adjusted" by some unspecified means. Also in Appendix D, "Transport Through the Containment", we feel that recalculating a result using different identified inputs is acceptable but post-calculation adjustment of the results of someone elses' calculation is a poor approach. The reasons for adjustments of this sort should be clearly set forth. We suspect that these two cases are not isolated instances.

APR 5 1982

continued...

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We understand that the conditions for the accident scenarios are based on the outputs from the MARCH Code (a 1978 version). There is a lack of confidence in the reactor safety community in the output of this code. For example, on May 21-22, 1981, a review meeting on the MARCH Code was held before the ACRS subcommittee on Class 9 accidents. A very large number of deficiencies in the code were pointed out at that time. It is therefore our opinion that this study should be redone using a more satisfactory procedure for the analysis of the thermal hydraulic response of the reactor core, the primary coolant system, etc., in typical accident sequences. It is our understanding that such codes are close to being available (for the present, we would recommend hand calculations plus the use of an updated version of MARCH). We feel that this is a particularly important point since we have good reason to believe that MARCH is predicting temperatures, pressures, flowrates, aerosol concentrations, etc., which are too severe and that therefore the reactor safety programs are being moved into regimes in which the conditions are too extreme.

Probably related to the extreme nature of the MARCH Code predictions is the assumption of an "entirely dry environment" in the primary containment system and in the reactor containment building in some of the accident sequences. In a large BWR there can be approximately 200,000 gals. in the primary system and 500,000 to 1,000,000 gals. in the suppression pool. In a large PWR there can be about 100,000 gals. in the primary system. To be able to eliminate so large an amount of water seems unrealistic. The burden of proof that this situation can occur should rest with the authors of the ORNL documents.

We suggest that the check valve "V" accident scenario be eliminated from the study since this accident scenario is no longer a dominant risk due to NRC administrative action designed to significantly lower the probability of failure on demand by these components.

The inputs and assumptions used in this study are claimed to be "realistic" rather than "conservative." However, this may not be the case. For example, it was stated in one of the ORNL viewgraphs that "iodine should be considered to be present as elemental iodine or as iodide, depending on which results in the larger consequences." It would appear that the philosophy of realistic calculations has not been thoroughly adopted. Therefore, when the complete sample calculations are presented it will be important that they be reviewed with this aspect in mind.

In conclusion, it may be helpful to point out that the Task Force which prepared NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power plants," (December 1978) concluded in Appendix III, Page 3 that "The Task Force believes that it is not appropriate to develop specific plans for the most severe and improbable Class 9 events." This philosophy does not seem to be consistent with the accident scenarios studied in the ORNL documents.

continued...

EPRI will be pleased to review the revised ORNL/TM-8275 and ORNL/TM-8274 (NUREG/CR-2639) when the complete calcualtions are available for us to study.

Sincerely yours,

Richard C. Vogel

Richard C. Vogel
Nuclear Power Division

RCV:ht

cc: C. Starr
F. Culler
J. Taylor
W. B. Loewenstein
D. Rossin
I. B. Wall

DRAFT

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~~C-17~~

Comments on Draft NUREG/CR-2629 (ORNL/TM-8274)

(Authors: S.J. Niemczyk and L. M. McDowell-Boyer)

Reviewer: S.K. Loyalka, Peer Review Meeting
(March 18, 1982), USNRC

The stated purpose of the work presented in these reports was to review some assumptions and recommend two accident spectra (for equipment qualification and emergency planning) for regulatory guidance related to light water reactors. Since the technology (the mechanistic approach) for reliable estimations of the source term is far from fully developed, the authors have chosen a semi-empirical approach. The basic components of this approach are the following:

(1) Accident sequences, and the probabilities of their occurrence are obtained from the Reactor Safety Study (RSS) and the subsequent Reactor Safety Study Methodology Applications Program (RSSMAP).

(2) Some "assumptions" and "procedures" (physical and computational models) are used to estimate the release fractions.

(3) Results of (1) and (2) above are combined to develop two "accident spectra" for the interim source term assumptions.

I do not think that for the purpose of the present work, the items (1) or (3) are in much dispute. The item (2), however, has been addressed in the Technical Bases Report (NUREG-0772), the RSSMAP studies, and the technical literature since the RSS, and has also been developed further in the present work. My specific comments, on the clarity of the report(s), its technical content, and the extent to which it has met its stated objectives are given below.

1. Clarity of the Report(s):

Considerable thought seems to have gone into the organization of the report. Chapter I provides a very clear description of the background, the purposes and objectives, the scope, the approach, and the organization of the report.

Chapters 2-6 are also well organized, and very informative. Appendices provide details which are appropriately separated from the main text.

The report is remarkably free of typographical errors (I could find only a few oversights). I think the words "unfortunately", "fortunately", "regretfully", "more exactly", have been overused throughout the text, and the authors may wish to minimize such usage. Also, references to unpublished works (private communications) should be minimized.

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2. Technical Content:

(i) The report contains excellent reviews and detailed descriptions of the natural and the engineered processes that are likely to affect the release fractions. Many of the tables are quite new, and very informative.

(ii) "Assumptions and Procedures", (Chapter 3 and Appendices B-D) put forth some simple and plausible physical and computational models. The approach is semi-empirical, and there is of necessity a strong "contraction of description". For example, relating the fate of any aerosol particle to the fate of the total aerosol mass may not be "realistically conservative" in that smaller, lighter particles are likely to remain suspended for a considerably longer period of time than the larger, heavier particles which may have a stronger correlation to the total mass. I think there should be some discussion of the "assumptions" themselves, their plausibility, and limitations.

(iii) "Limitations" and "gaps" sections of various chapters, while generally correct and very useful, could be expanded to emphasize the present lack of understanding of the basic release processes, the chemical and physical behavior of the released products, the complexity and synergism of the processes, etc. It should be pointed out that the processes and phenomena are highly non-linear and dynamic, and that development of verified predictive tools may require not only an understanding of the effect of the first order, but also the seemingly second and third order affects which in synergism with the former, can have greater importance than presently realized. For example, preliminary results suggest that inclusion of diffusiophoresis and thermophoresis with gravitational motion, can alter substantially the estimates for agglomeration rates from the estimates obtained by the consideration of gravitational motion alone.

(iv) Chapter 3, Appendix B: Estimation of release fractions, as a function of temperature (and time) appears meaningful. The discussions of pages B-65 and B-66 are informative.

(v) Chapter 3, Appendix C: The authors have carried out a careful evaluation of the processes that affect the release fractions in the PCS. Perhaps, Table C.1 should also include homogeneous nucleation and fragmentation as two possible processes of some consequence.

(vi) Chapter 3 and Appendix D: The authors have provided an excellent description of their work and the limitations of it. I particularly liked the tables D.1-D.5. I think Table D.1 perhaps should include homogeneous nucleation as a possible process. Also, it should be pointed out that many of the processes (Table D.1), in synergism, could increase or decrease the effects of the individual processes on total aerosol mass in important ways (for example, agglomeration and condensation could influence each other).

(vii) Chapters 4-5 and Appendix E: Both Chapter 4 and 5 are well written and to the point. The appendix provides a good description of the technical details. The results of the present work have been compared with previous estimates (RSS, RSSMAP, TID 14844, Reg. 1.89) in a clear and forthright manner. Uncertainties in the estimations and recommendations have been discussed quite clearly.

(viii) Chapter 6 and Appendix E (pages E-11 - E-27)

I find myself in considerable agreement with the views expressed in Appendix E (pages E-11 - E-27). Clearly, good engineering is nothing if it is not a painstaking attention to the detail. Considerable progress has been made in the past few years towards delineation and understanding of the basic processes that could be important in source quantifications. The technology for such quantification is, however, still not very good. In the absence of good technology, rapid generalizations based on slender evidence should be avoided.

I think the discussions of Chapter 6 should be more in conformity with those of Appendix E. It appears to me that Chapter 6 concentrates on the Technical Bases Report (NUREG-0722) more than necessary. What is needed I believe is a brief discussion of the state of the art, the inherent limitations of the present technology and how these limitations may be alleviated in a systematic and timely manner. In view of the complexity of the problems, research must provide answers to the questions of the levels of "contractions of descriptions" that may be possible through experimentally validated theoretical and computational procedures. The authors might wish to consider the following suggestions for the Section 6.2:

(a) Recommendations 1 to 3 should be clarified, and made more to the point.

(b) Suggestions 4 and 5 should be clarified. What is necessary are the resources and the commitments towards developing the technology in sustained, unbiased, competent, innovative and purposeful manners. The task here would benefit through encouragement of both confirmatory and exploratory research, and interaction of all serious researchers.

3. Realization of the Objectives:

The authors have obtained results that appear reasonable. The results are based on the state of the art, which is not very good. The uncertainities in the results are rather large, and not quite quantifiable at this time. Better quantified results would certainly be obtained in the future. The authors have provided some important insights into the shortcomings of the present state of the art, and on how one might alleviate these shortcomings in the future.

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Offshore Power Systems
[illegible] Jacksonville Florida 32211
Telex 568403

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March 30, 1982

Mr. L. G. Hulman, Chief
Accident Evaluations Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop P802
Washington, D. C. 20555

Dear Mr. Hulman:

I am sorry that other commitments prevented my attending the NRC Peer Review of the ORNL Draft Report on "Interim Source Term Assumptions for Emergency Planning and Equipment Qualifications." I have, however, prepared comments as you requested the peer reviewers do and they are submitted for your consideration. Comments are attached which address four areas, they being technical, editorial, policy implications and response to the specific NRC questions proposed at the Peer Review Session.

In an overall sense, the source term values calculated in the two ORNL reports are not very different than those of the Reactor Safety Studies (RSS) or of NUREG-0772. Values from these three sources are based on models derived basically from the same source of data so the similarity is to be expected. However, radioactivity release values for core melt accidents calculated with existing methods are generally acknowledged by the technical community to be conservatively high.

With respect to emergency planning, I believe application of the source values contained in the ORNL reports would be premature. Policy decisions regarding the way the safety goal is to be applied in emergency planning and the severity of the accident to be considered for emergency planning purposes (the most severe, the risk dominant, other) are needed before appropriate accident sequences can be selected. Further estimates of realistic source terms (or of the degree of conservatism in existing values) are needed before radiological release values for emergency planning purposes are selected.

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March 30, 1982
Mr. L. G. Hulman

With respect to sources for equipment qualification, values suggested in the Oak Ridge report do not imply that any large upward adjustment in radiological exposure values for equipment qualification is justified. In view of the conservatism in values now being applied, upward adjustment of the sources for calculating exposure values does not seem appropriate at this time.

Sincerely,

D. H. Walker mdc

D. H. Walker, Chief Engineer
Nuclear Engineering

/aw

Enclosures

cc: W. E. Bennett - Consolidated Edison
J. F. Davis - Power Authority of State of N.Y.
E. Fuller - TEC
P. Rahe - W-NTD
G. Rymer - W-NTD
N. J. Liparulo - W-NTD
J. Siegel - AIF
R. C. Vogle - EPRI
M. A. Capo
R. J. Cooney
P. B. Haga

PST-NE-528
Enclosure

Technical Comments on ORNL Reports NUREG/CR-2629 and ORNL/TM-8275

1. The source terms presented in the ORNL reports for the various PWR and BWR accident sequences are calculated employing generally the methods and models of the Reactor Safety Study (RSS) or those presented in NUREG-0772. Calculated values are not very different from those of the RSS (with few exceptions they differ by less than a factor of 2 for radionuclides contributing significantly to consequences). Values estimated in NUREG-0772 were also generally about the same.

   The approach of the RSS, NUREG-0772 and the 2 ORNL reports is generally acknowledged in the technical community to produce conservatively high source term values for specific accident sequences, so the values are "safe" in a licensing sense. Estimates of the degree of conservatism vary ranging from a factor of 2 or 3 to several decades. However, in spite of the acknowledged conservatism, existing data is just not sufficient to provide a basis for new models which would lead to significant downward adjustment of the source values at this time. There is then the dilema of source terms generally acknowledged to be conservatively high without an adequate experimental basis for improved models to calculate the degree of downward adjustment. The technical comments are primarily based on this "state of the art." I believe it would be appropriate for the ORNL reports to also acknowledge this state of affairs.

2. In the 2nd sentence of the report summary (NUREG/CR-2629), the purpose is said to be to develop more realistic source term assumptions. The report does not accomplish that purpose and I doubt that such an objective can be accomplished by applying analytical methods available today.

   With respect to the methods applied to obtain the results, traceability within the report is poor. Its difficult to determine which methods the authors applied to obtain particular results. However, in a technical sense, it probably doesn't make much difference since calculations by various investigators all seem to give about the same results. I think the results are not very sensitive to which of the generally accepted methods are applied and, therefore, believe the methods employed to arrive at the values in the ORNL reports are acceptble as long as one recognizes that existing methods all yield source term estimates which are conservatively high by a large factor.

3. The one method we have applied for incorporating realism into the source term values is the quantification of expert judgement in terms of probability distributions. This method was applied and is reported in the Zion and Indian Point Probabilistic Safety Studies. There are plans to apply the approach to a broader spectrum of accident sequences and plant types as part of IDCOR Task 11.3. NRC should consider incorporating the results of such a task into NRC accident source terms for emergency planning purposes and PRA evaluations.

4. I do not agree with the last two paragraphs of Section 5.6, Page 11. Calculated source term estimates for the accident sequences are conservative; therefore, reductions in their magnitude can be expected as additional data becomes available. It is incumbent upon us to attempt to quantify the extent of that conservatism before the source terms are applied for emergency planning purposes. It is also necessary to address policy questions like how the proposed safety goal is to be applied for emergency planning considerations before accident classes and associated source terms are selected for regulatory guidance for emergency planning. It is much too early to adopt values of Table 5.7 for that purpose.

   Also, current industry studies are indicating probabilities for accident sequences that involve both core melt and loss of containment function to be significantly lower than those listed in Table 2.3 (largely as a result of more careful evaluation of containment failure modes). These lower probabilities need to be factored into emergency planning considerations.

5. Section 3.3.2, page 8 (NUREG/CR-2629)

   The recommendation to "not consider permanent retention of radioactivity in the reactor coolant system" does not represent a realistic approach as claimed in the document summary.

6. Section 3.6, page 10

   Some studies have indicated aerosol concentrations in containment may be significantly higher than those used in the CSE tests. It is not appropriate to limit concentrations to the low levels of the CSE tests at this time. Enhanced removal by increased mass loading may prove to be appropriate for source term estimates.

7. Table 4.2 (NUREG/CR-2629)

   Inclusion of all of noble gases in the water borne source is not appropriate.

8. Section 4.6, 2nd Paragraph (NUREG/CR-2629)

   I do not understand the need for an upward adjustment factor for source term values that already represent complete release of the core inventory to the containment. Certainly a complete release assumption represents adequate conservatism in and of itself.

9. Tables 5.2 and 5.3 (NUREG/CR-2629)

   One has the feeling there should be some correlation between assumptions in Table 5.2 and escape fraction estimates of Table 5.3. It is not obvious to the reader there is any correlation and the text doesn't help. What does footnote (a) on Table 5.3 refer to? Escape fraction estimates in Table 5.2 need to be acknowledged to be estimates based on conservative judgement.

EDITORIAL COMMENTS

1) The report suffers from an overdose of prose. One finds the same or slightly modified writeups occurring a number of times. I think the approach imposes unnecessarily on both the reader and the eventual user. Like good style, conciseness is a virtue to be practiced in writing a report.

2) The report lacks crisp explanation of how results are obtained. As a result, traceability of assumptions and calculational approach applied to obtain the results is poor.

POLICY IMPLICATIONS

1) It seems to me that there are a number of policy questions that need to be addressed before source terms for emergency planning purposes are selected. Questions that need to be addressed include:

   a. How will the safety goal be utilized or applied in selecting the accident sequences upon which emergency planning is based?

   b. PRA analysis indicate there is a class of very low probability, high consequence accident sequences which are not usually significant contributors to overall risk. Are these an appropriate class of accident sequences upon which to base emergency planning considerations or are the accident sequences which are the dominant contributors to risk more appropriate? (To me, the latter seems a more appropriate choice).

   c. Is it appropriate to use source term estimates known to be conservatively high or to use realistic values for emergency planning considerations?

   Decisions on such questions will have a marked influence on the approach used for estimating accident source terms. Until such policy issues are addressed, I do not believe source term selection is an appropriate exercise for emergency planning purposes.

2) With respect to qualification of equipment for operation in the post accident containment environments, generally past practice has been to utilize radiation levels calculated from Reg. Guides 1.3 and 1.4 sources (100% noble gases, 50% halogens, 1% other fission products). The recently issued for comment Rev. 1 of Reg. Guide 1.89 increased the basic source terms by including 50% of the core inventory of cesium in the sump water. The energy release curves presented at the March 18, meeting suggest that iodine, noble gases, and cesium (if properly plotted) contribute about the same magnitude to the source,

with iodine being the major contributor after a very short time period and out to 280 hours. The ORNL report suggests increasing the cesium release to the source term to 100% of the core inventory. At most, this would increase the dose calculation by a factor of 2 to 3.

In view of the known conservatively high nature of source terms, I believe careful consideration should be given to any upward adjustment of source terms for equipment qualification particularly when such an adjustment would be at most a factor of 2 or 3. Cost or requalification for existing equipment is substantial; justification of new and higher source values based on conservatively calculated source values for the most severe possible sequences is hardly appropriate and I recommend equipment qualification dose levels not to be changed at this time.

NRC QUESTIONS

Five questions were posed at the March 18, peer review meeting. Responses follow:

1) Need for and timing of new source term assumptions. See Policy Implication discussion preceding. I believe fundamental policy decisions are needed to provide a basis for new source term assumptions. Once a safety goal and the way it is to be implemented are defined, source term for emergency planning purposes need to be carefully examined.

2) Were the right sequences defined?

   a. For Emergency Planning

      An adequate basis does not exist for selecting the "right" sequences for emergency planning purposes. Policy decisions are needed before such sequences can be selected; see preceding discussion.

   b. For equipment qualification

      For the present, current assumptions are adequate; see previous discussion.

3) Are calculational procedures used adequate for the job?

   a. Release from the fuel

      Calculated release fractions are probably conservatively high but reasonable.

   b. Primary system retention and transport

      Existing calculational methods are not adequate. The assumption of no retention is conservative.

c. Containment processes

Existing calculational methods account for important processes but yield conservatively high source term release values.

4) Are the results of the sample calculations reasonable?

See Technical Comments section. Values calculated are conservatively high by significant factors. A complete "from beginning to end" sample calculation needs to be in the report. The techniques employed are presented in a fragmented fashion, so it is difficult to critique the overall method employed to generate the source terms.

5) Is there new information since NUREG-0772 which would substantially change source term values?

No. However, that does not mean source values are adequate or appropriate for equipment qualification or emergency planning.

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Battelle
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March 22, 1982

Ms. Sue Gull
Oak Ridge National Laboratory
Box X-10
Oak Ridge, Tennessee 37830

Dear Sue:

You and Laura are to be congratulated on putting together a fine report and an excellent presentation in Bethesda last week.

I offer the following comments regarding aerosol transport in the PCS for your consideration:

- As I mentioned to you at the review meeting, I think that the inclusion of a set of sample calculations used to arrive at the final release fractions presented would be helpful to illustrate the reasoning used and make the assumptions involved more explicit.
- Inclusion of a figure showing aerosol mass released from the core and from the PCS as a function of time along with an indication of the time of core slumping and RPV failure. This would permit the uncertainties in the PCS escape fraction due to emissions from the molten pool to be assessed.
- The footnote to Table C.4 should be changed so that the initial number concentrations are $10^{10}/cm^3$, $10^9/cm^3$, and $10^8/cm^3$ (This was an error in the table I sent you. Sorry!).
- CsI is expressed as CI in a couple of places in Appendix C.

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- In the assumptions in Section C.4.2.1 it's not clear whether CsI and Te are to be treated as vapors or aerosols. I didn't think there was enough justification in the previous discussion to assume no vapor retention in the PCS.

I look forward to seeing the final product of your efforts.

Yours truly,

Mike Kuhlman

Michael R. Kuhlman
Physico-Chemical Systems Section

MRK:drr

cc: L. G. Hulman (NRC)

COMMENTS ON ORNL/TM-8275 ORNL/TM-8274(NUREG/CR-2639) AND ACCOMPANYING APPENDICES

R. C. VOGEL

EPRI

(Comments Submitted at the Peer Review Meeting)

- The review of these lengthy documents was rushed since we had only four working days.

- The review is made difficult by the lack of visible "beginning-to-end" calculations for several sample accident scenarios. If a meaningful review is to be undertaken, the following information should be available:

  (1) Identification of the codes used, the input to the codes, the underlying principles involved in the codes and the assumptions implicit in the codes.

  (2) A discussion of the validity of the input data is needed. It is necessary that these data be clearly set forth along with their inadequacies. It is noted, for example, that the release rate data of NUREG-0772 are in part used but are "adjusted." We feel that the release rate data of 0772 are indeed poorly supported by experiments. However, the rationale for changing the data without further experimental information should be thoroughly laid out.

- It is noted that the MARCH code is basic to the study. This is believed to have been a good code at the time it was developed. However, currently it is inadequate for current applications.

In a recent meeting (Feb. 24, 1982) on the MARCH-2 code sponsored by NRC a table of 43 planned improvements on MARCH-2 was presented. It is assumed that for this study under review an earlier, less-refined version of MARCH was used. It is therefore generally recognized by the reactor safety community that MARCH needs considerable improvement. For example, some of the problems are as follows:

(1) Lack of treatment of hydrogen diffusion limited metal-water reaction.

(2) Heat transfer from partially covered nodes inadequately treated.

(3) Apparent lack of mass and energy balances.

(4) The radiation heat transfer models are sometimes unnecessarily crude. For example, there is an improper use of view factors between radiating surfaces, upper internal structures absorbing radiation heat without changing structure temperature, etc.

(5) The computed water properties were inadequate except for saturation values.

It is our suspicion that MARCH is predicting temperature, pressure and aerosol concentrations which are too severe and therefore moving the reactor safety programs into regimes in which the conditions are too extreme. This possibility should be checked by calculations using better codes.

- It is believed that the aerosol treatment is inadequate. Specific examples are the following:

  (1) The high concentration ($>$200g/m^3) aerosols agglomerate to very large sizes in a few seconds. Thus, the length of residence time in the RPS is of no consequence, since once the agglomeration has occurred, the fallout location is immaterial.

  (2) The resuspension aerosols that have fallen out previously have a size distribution shifted to much larger sizes due to agglomeration with other fallout aerosols on the wall.

  (3) As the steam in the containment condenses on the walls and machinery (especially the containment coolers), it creates a convective flow which drags the aerosols to the water film on the walls. This is the most potent air cleaning mechanism in the RCB and has been demonstrated at various labs including ORNL(*). However, CORRAL contains a correlation based on the CSE experiments which were performed under isothermal conditions that lacked the convective flow. Such effects were observed in the PRTR accident (Hanford-PNL)[+].

  (4) The reports continually refer to "dry" primary coolant system and "dry" containment buildings with possible oxygen containing atmospheres, but it is not explained how these conditions are obtained. These two appear to be inconsistent in any one accident scenario. An explanation of where

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(*) See: T.S. Kress & M.L. Tobias, "LMFBR Aerosol Release & Transport Program Quarterly Progress Report for July-September 1981, NUREG/CR-2299, Vol. 3, (1982) the water goes in a double dry scenario is required.

(5) The aerosols escape the containment as if they were gases, when in fact they will deposit in cracks. (For a RCB leaking gas at 0.1% per day, < 1 mg of aerosol will escape)**

+R. W. Perkins, C. W. Thomas and W. B. Selken, "Fission Product Aerosol Behavior in the PRTR Fuel Rod Failure of Sept. 29, 1965," BNWL-SA-668 (1965)

**H. A. Morewitz, "Aerosol Leakage from Containment Buildings," Health Physics , (1982)

- Specific Comments on Appendix A, "Accident Descriptions and Terminology", are as follows:

  (1) "Entirely dry environment" in PCS probably doesn't exist. The same is true in RCB. With in excess of 200,000 gal of water inventory in primary system a completely dry system or containment is unrealistic. (section A.2 and other places)

  (2) Even for hot leg breaks, attenuation in PCS may be substantial when the mass temperature and geometry of PV internals are properly considered. (section A.2.2 footnote)

  (3) Fission products passing through a boiling suppression pool would still be scrubbed fairly effectively (vapor/liquid partition), although not as effectively as in a cold pool. (page A-9 TW-γ, γ' and other places)

  (4) Ice-condenser PWR - the atmosphere during any accident would produce a wet and steamy containment building, which would tend to mitigate the effects of a H_2 burn and to attenuate fission products. Same is true for BWR and other PWR containments. (page A-10, section A.3.3)

  (5) Check valve "V" accident scenario is no longer a dominant risk scenario due to recent NRC action which is designed to significantly lower the probability of failure on demand by these components. Since it is not a large contributor to risk, it should no longer be considered. (page A-7, section A.3.1)

- Specific Comments on Appendix C, "Transport in Primary Coolant System," are as follows:

(1) It is not only the attenuation of fission products in the PCS that is important, but also the fact that fission products which manage to survive the PCS are changed in character in a way that allows rapid attenuation once they get into the RCB; e.g., vapor forms an aerosol, ages, partially agglomerates, and even though it may not deposit in PCS it can rapidly fall out in containment. (page C-3, general comment)

(2) Any resuspension of f.p. will not be very important because the resuspended material will be agglomerated particles of large diameter and mass, and therefore will not be a persistent aerosol. (page C-4, general comment)

(3) The temperatures of the PCS are much lower (by up to a factor of 2) than the MARCH code predicts. This is shown by ANCHAR code which has been benchmarked on TMI. Therefore, steam condensation and vapor deposition is likely to be important in PCS. (Table C.1, general comment)

(4) Materials released from the core do not get transported instantly to the upper plenum. At the interface of the core melt region high temperature gradients exist driving aerosols to nearby structure, and plateout will be important. In addition, the upper plenum is highly complicated geometrically. Although the lower part may be relatively hot, the upper internals will be relatively cool because of their mass and heat capacity and because of the thermal radiation shielding provided by the lower plate. (page C-11)

(5) The report places great emphasis on elemental iodine. There is no demonstrated mechanism for producing I_2, and it is only likely to exist in negligible quantities, if at all. (page C-11)

(6) If elemental iodine were to exist, reaction with PCS surfaces would be quite important, especially with upper internals. (Recent experiments[+] have shown that the deposition velocity of I_2 on stainless steel increases sharply above 600°C metal temperature). (page C-12 footnote)

(7) Page C-15 lists a number of reasons why the Technical Bases Report supposedly gave low release fractions. We agree that that report gave incorrect results, but the results probably over-estimated rather than under-estimated the transmission of fission products through the PCS. For Example: point 1: local concentrations are important, the total mass of aerosols should not be necessarily uniformly distributed throughout the RPV. The upper plenum is "subcompartmentalized," and the temperatures low at the top of this region. In general, the role of the upper internals has been neither correctly taken into account by the Technical Bases Report, nor correctly interpreted by the current report.

Similar analysis pertains to the 5 other points. Net effect would be that the escape fractions from the PCS, very high to start with, are made even higher by the subject report, in spite of the fact that they should be lowered. (page C-15)

[+] S. L. Nicolosi and P. Baybutt, "Vapor Deposition Velocity Measurements and Correlations for I_2 and C_sI," Battelle Columbus Lab Report #BMI-2091 (1982)

(8) Since surface temperatures in most of the PCS are unlikely to exceed 800°C (and in most instances will remain below 500°C) revolatilization is not likely to be an important escape mechanism, even though some migration may be expected in some quite local regions very close to the melt zone. (page C-18)

(9) High steam flow rates are only likely during the blowdown of the system, before the evolution of volatile fission products, or when the molten core drips into water in the lower plenum, long after the volatiles have been released and have already escaped or been attenuated. During boiloff, system temperatures would be quite low. Delayed operation of ECCS, would cool the upper internals prior to the core, effectively quenching and attenuating the aerosols. (page C-20, section C.4.1.3)

(10) No understanding of the importance of "collective effects" in high density aerosols is evidenced in this table or in the discussion. (page 22, Table C.4)

(11) What elemental halogens exist? Section seems to dismiss plateout as a potentially important mechanism. (page C-28, section C.4.2.1)

(12) No understanding of the hydroscopic nature of important vapors, such as CsI, has been demonstrated in this section. The scrubbing of such vapors and aerosols is greatly enhanced and is significant even in saturated pools. (page C-30, section C.4.2.4)

(13) Table is a subjective estimate. All releases from the PCS (except noble gases) are much too high. The ranges given are not reasonable. (page C-31, Table C.5, Table C.6)

(14) Table correctly identifies many problems associated with estimating releases. However, correcting the deficiencies would greatly reduce the release fractions, not increase them. (page C-34, Table C.7)

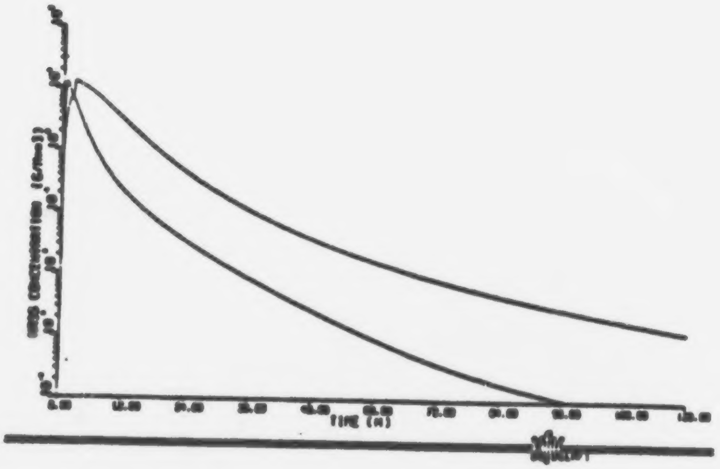
- Specific Comments on Appendix D, "Transport Through the Containment," are as follows:

  (1) The document states (p. D-6) "Aerosol removal is also augmented by another condition which would be typical of many meltdown accidents, that is, a condensing steam environment." CORRAL was derived from the CSE experiments which were isothermal. Isothermal experiments will not show the effect referred to on p. D-6. Additionally, this effect can be expected to be the dominant scrubbing mechanism in containment. (D-6 & D-14)

  (2) Comparison of HAARM-3, CORRAL-2, NAUA-4 and QUICK. Attached are figures showing NAUA wet and dry calculations from the NAUA documentation. These attached plots show significant differences between the wet and dry calculations unlike figure D-1. The discrepancy with figure D-1 is puzzling. It does not appear that these calculations shown in figure D-1 were done for the identical case for all codes. The CSNI/SOAR No.1 on Nuclear Aerosols (1979) shows good agreement between HAARM, HAA, NAUA and other international aerosol codes. This document draws the opposite conclusion to p. D-19 that LMFBR codes are not applicable to LWR accidents. These codes can be applied to LWR accidents. The table D-5 overstates the abilities of CORRAL. It is misleading to claim in Table D.5 that CORRAL-2 handles Brownian Coagulation, Gravitational Coagulation, Turbulent Coagulation, Steam Condensation on particles, vapor sorption on particles, diffusiophoresis, and electrostatic interactions.

URA TRANSLATION NO. 1024

H. BUNZ, et al, "NAUA-MOD 3", KfK 3/54 (Sept 1981)



Influence of the Steam Condensation on the Aerosol Disintegration

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CORRAL-2 cannot extrapolate these phenomenae beyond the CSE data base. A careful examination of the applications in this study compared to these limitations would clarify the situation. In terms of predicting containment aerosol response in LWR containments, the HAA-4A code being developed by DOE is currently the most advanced in the U. S. Other codes in Table D.5 are all also much better at predicting aerosol response in containment than CORRAL. (D-22, Fig. D-1)

(3) The discussion of the leak path attenuation of aerosols (page 39) states "... because the correction is small and poorly described for large leak accidents..." and later for small leaks the aerosol could be attenuated but attenuation is ignored despite "...potentially large understated corrections for these accidents." Aerosol does not leak from the containment like a gas (implied assumption in this document). This effect is important even for large leaks. H. Morewitz has published a paper demonstrating aerosol blockage of a 30-inch pipe. To be realistic, this must be considered. (D38-39).

(4) The whole approach to adjusting CORRAL calculations after the fact is technically indefensible. Recalculating the number is an acceptable approach. Post-calculation massage of the numbers leads to difficulty. (D-46, D-48 & 49)

(5) These results are difficult to interpret. Does "Xe" include Kr? What species constitute "aerosols?" What is meant by "fraction of iodine present as elemental iodine which escapes?" How is this related to the core inventory of iodine? (D-52, Table D.7)

(6) It is stated that "because of the somewhat arbitrary nature of the definition and choice of the accident sequences..." "...problems with many of the detailed models...are not overwhelmingly important." This statement does not seem defensible. (D-54)

~~6-42~~

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THERMOCHEMISTRY OF FISSION PRODUCT IN STEAM & HYDROGEN

THE INFLUENCE OF VAPOR PHASE HYDROXIDES ON THE VAPOR PRESSURE OF FISSION PRODUCTS

PRESENTATION OF DANA A. POWERS

SANDIA NATIONAL LABORATORY

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VARIABLES IN THE THERMODYNAMIC CALCULATIONS

I). TEMPERATURE	500-3000 K
II). P_{H_2} / P_{H_2O}	10^{-3} TO 10^{2}
III). $P_{H_2} + P_{H_2O}$	1,5,15,100,150 ATMS.

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SPECIES IN THE ALKALINE EARTH SYSTEMS

I). STRONTIUM

SrO (c)
Sr (g)
SrO (g)
$SrOH$ (g)
$Sr(OH)_2$ (g)

II). BARIUM

BaO (c)
Ba (g)
BaO (g)
$BaOH$ (g)
$Ba(OH)_2$ (G)

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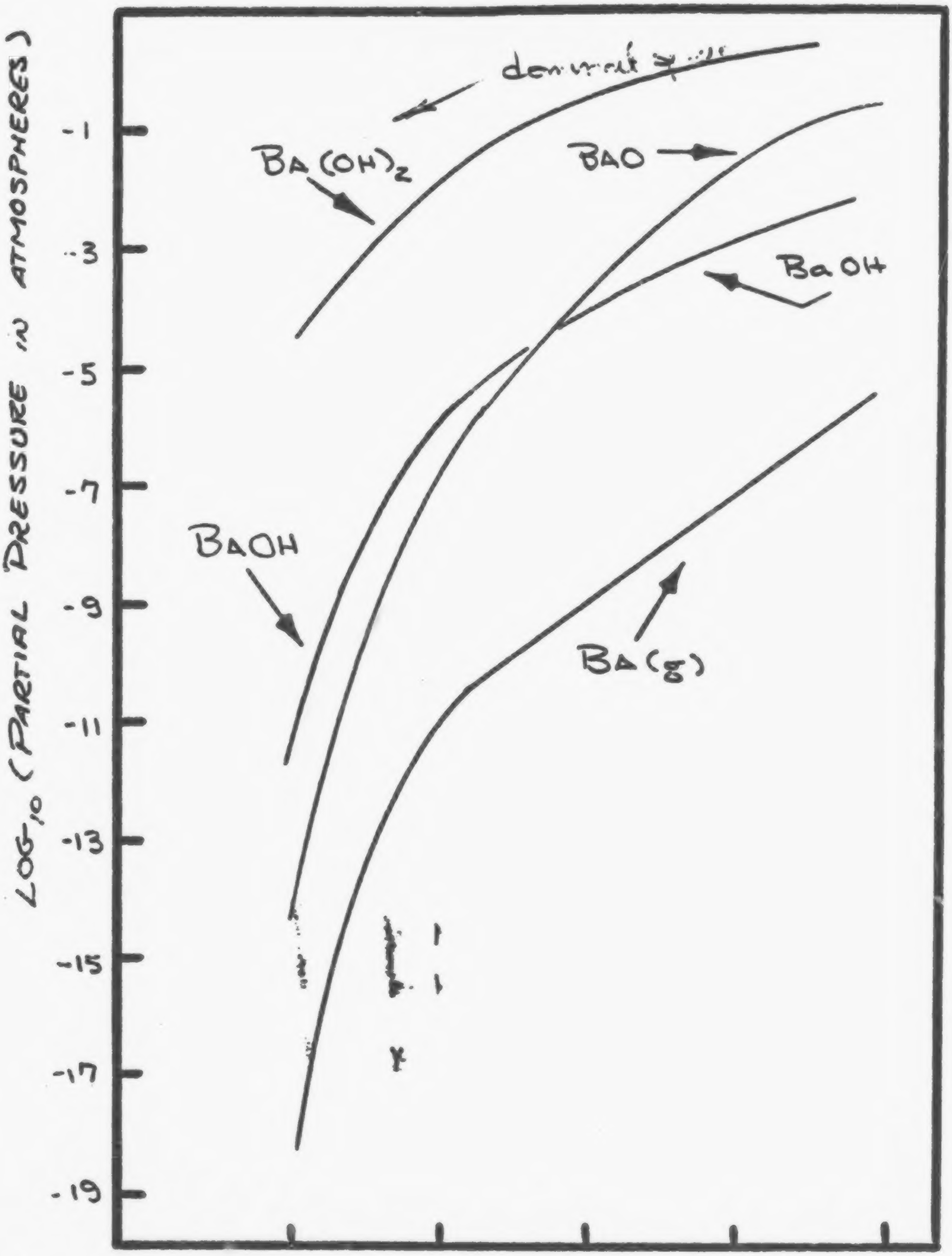
15 [illegible] ~~C-45~~

$H_2/H_2O = .1$



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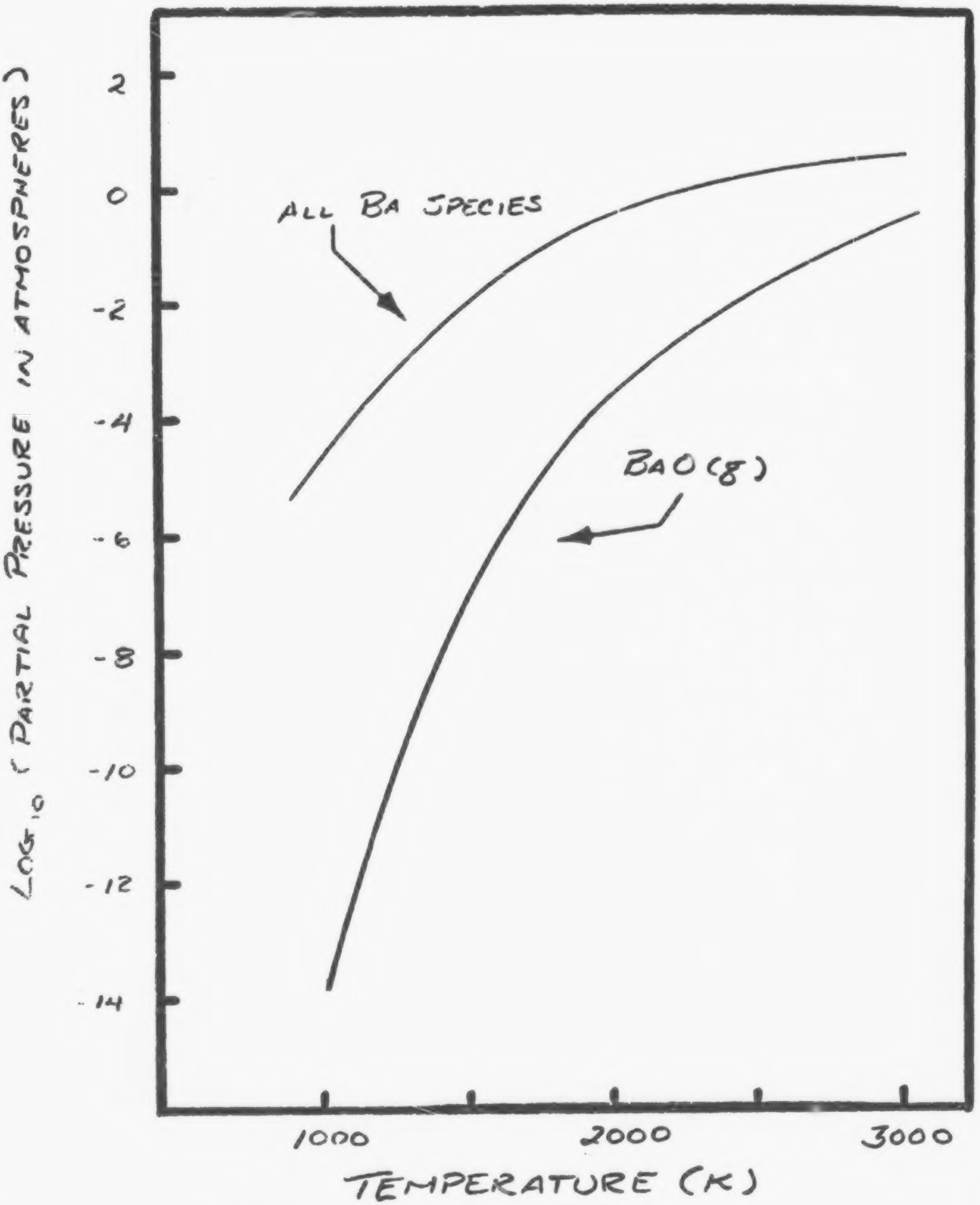

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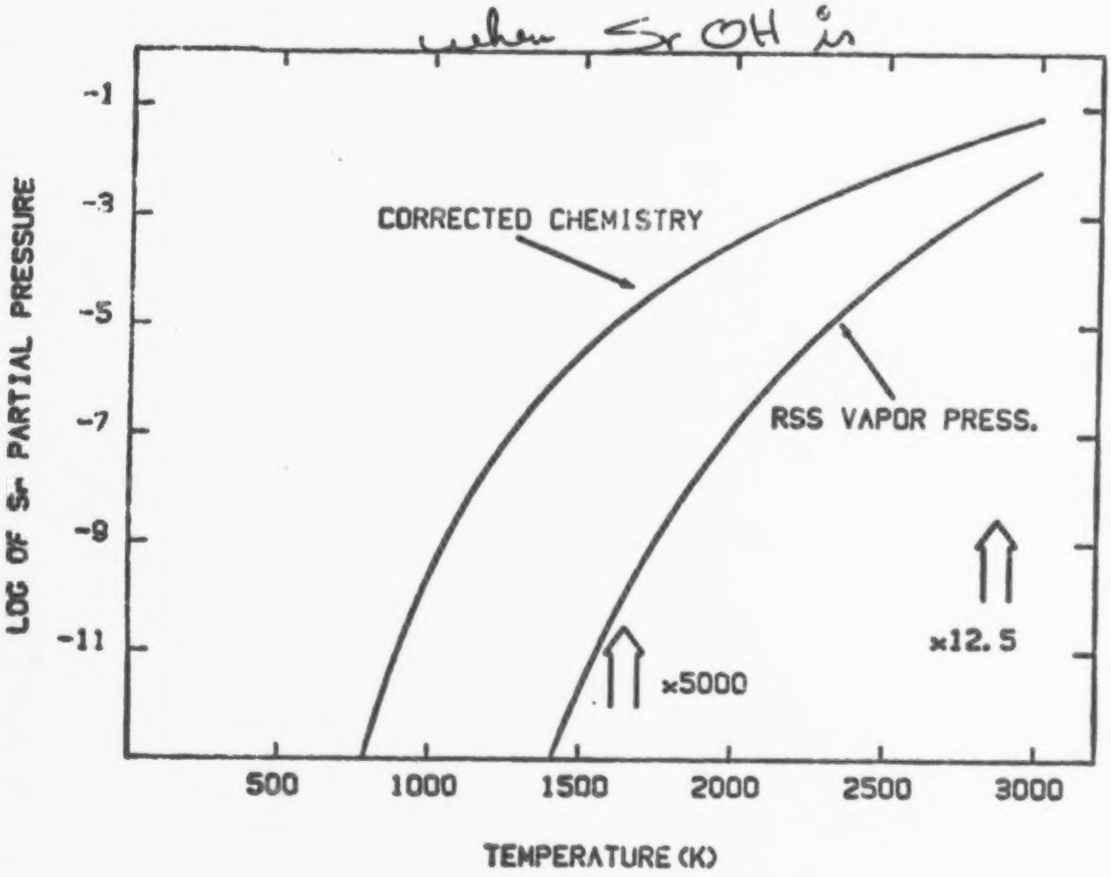
16D

FISSION PRODUCT RELEASE

Strontium
when SrOH is



with these numbers you will get order of
5-25% release of Sr

RELEASE	EARLY FATALITIES	EARLY INJURIES	LATENT CANCERS
RSS RELEASE	100%	100%	100%
7 X Sr RELEASE	141%	123%	106%
10 X Sr RELEASE	157%	137%	113%

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SPECIES CONSIDERED IN THE MOLYBDENUM SYSTEM

$Mo(c)$

$MoO_2(c)$

$MoO3(c)$

$Mo\ (G)$

$MoO\ (G)$

$MoO_2(G)$

$MoO_3(G)$

$(MoO_3)_N$ $N=1-5$

$MoOH\ (G)$

$Mo(OH)_2\ (G)$

$MoO_2(OH)_2\ (G)$

pure phase only - fuel not considered
molybdates not considered

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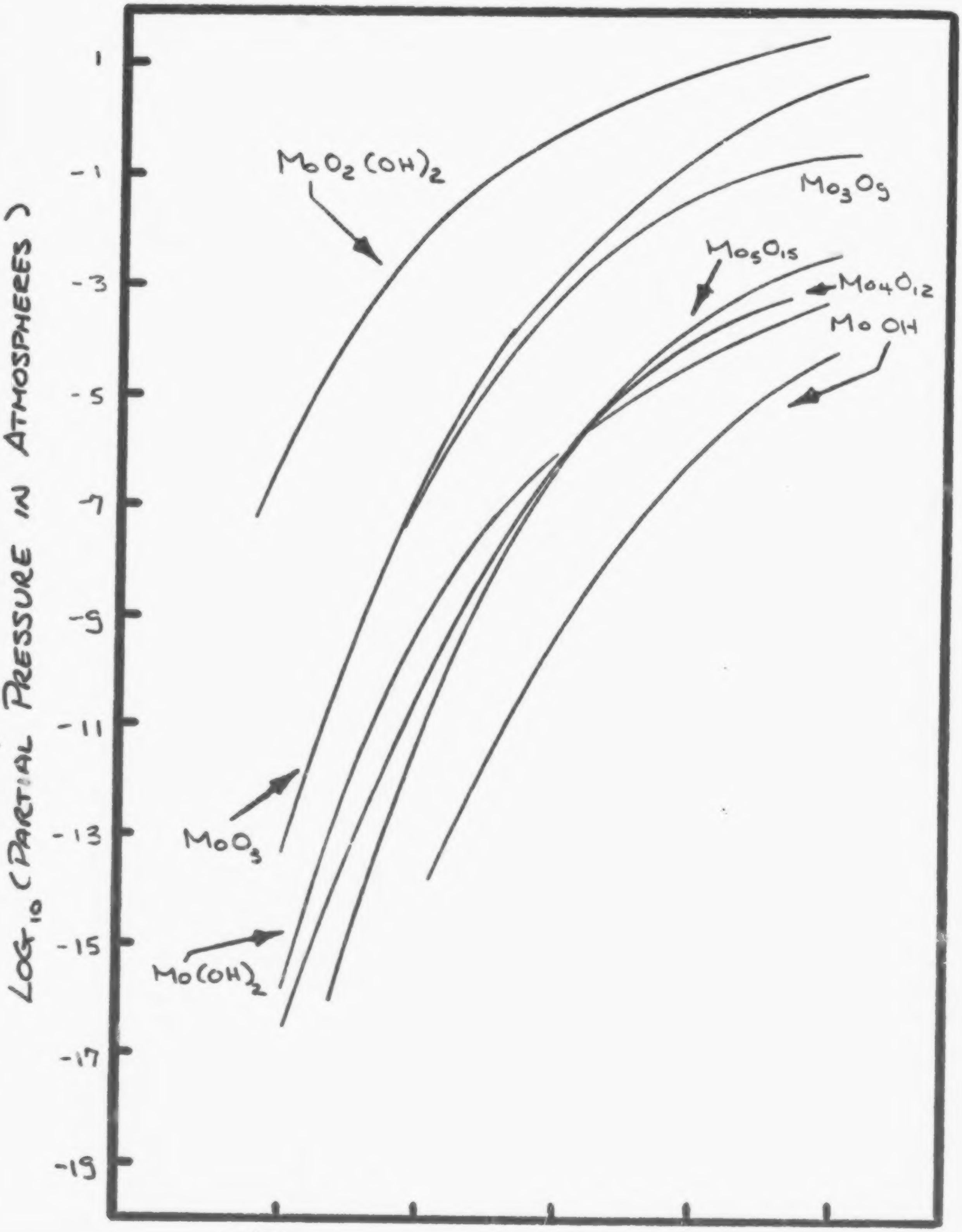
$\alpha \equiv H_2/H_2O$ 15atm total pressure ~~6.49~~



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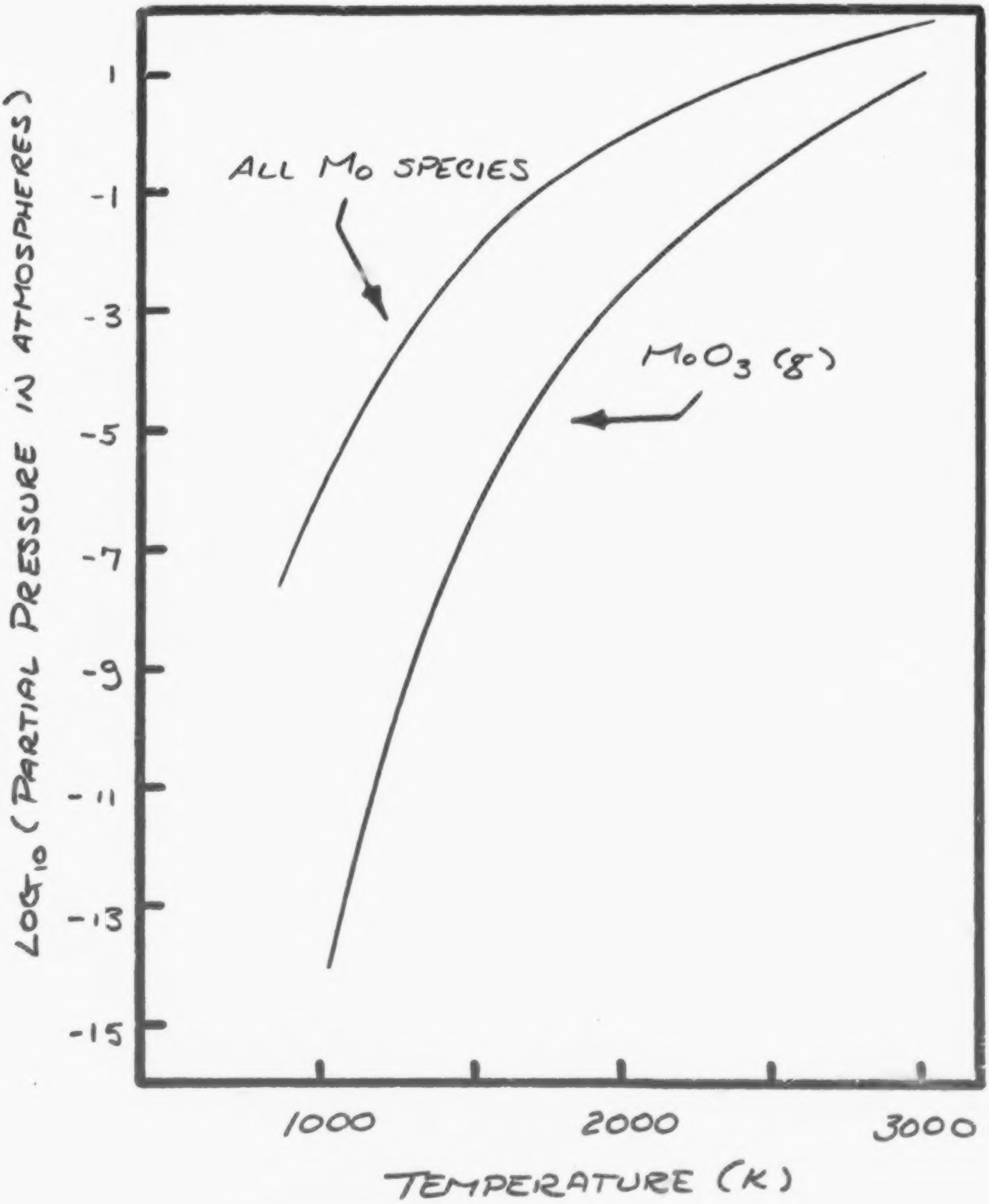

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method to

164 — guess other important G-57 hydroxide species

VAPOR PHASE HALIDES & HYDROXIDES CONSIDERED IN THE CORRELATION

CaX
SrX
BaX
MgX
FeX

NaX
KX
CsX
LiX
AlX

CaX_2
SrX_2
BaX_2
MgX_2
FeX_2

$AlOX$
BeX
BeX_2
BX_2
BX_3

BOX
BF_2X

$X = Cl^-, F^-, OH^-$

MoO_2Y_2
WO_2Y_2

$Y = Cl^-, OH^-$

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REVISED CORRELATIONS

I) BOND ENERGY

$$D_{M-OH} = (0.8953 \pm 0.03351) D_{M-F} + (12.65 \pm 1.20) X_M - 29.086$$

$$R^2 = 99.126\%$$

PROBABLE ERROR = 1.24 KCAL

X_M = ELECTRONEGATIVITY OF METAL ION

II) ENTROPY

$$S\left(M(OH)_N\right) = (1.065 \pm 0.055)\ S(MCl_N) - 5.4367$$

$$R^2 = 95.336\%$$

PROBABLE ERROR = 1.55 E.U.

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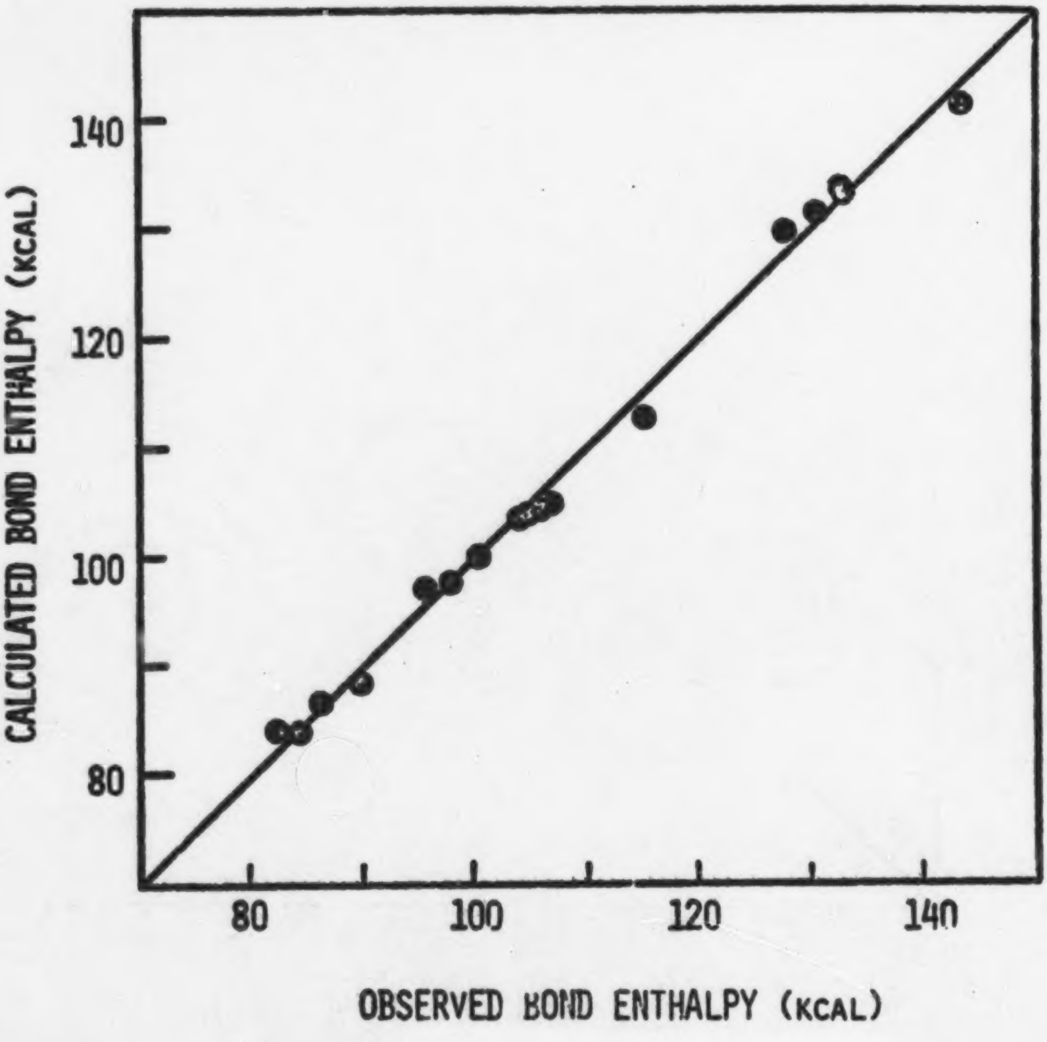
ENTHALPY CORRELATION

$D_{M-OH} = F(D_{M-F})$



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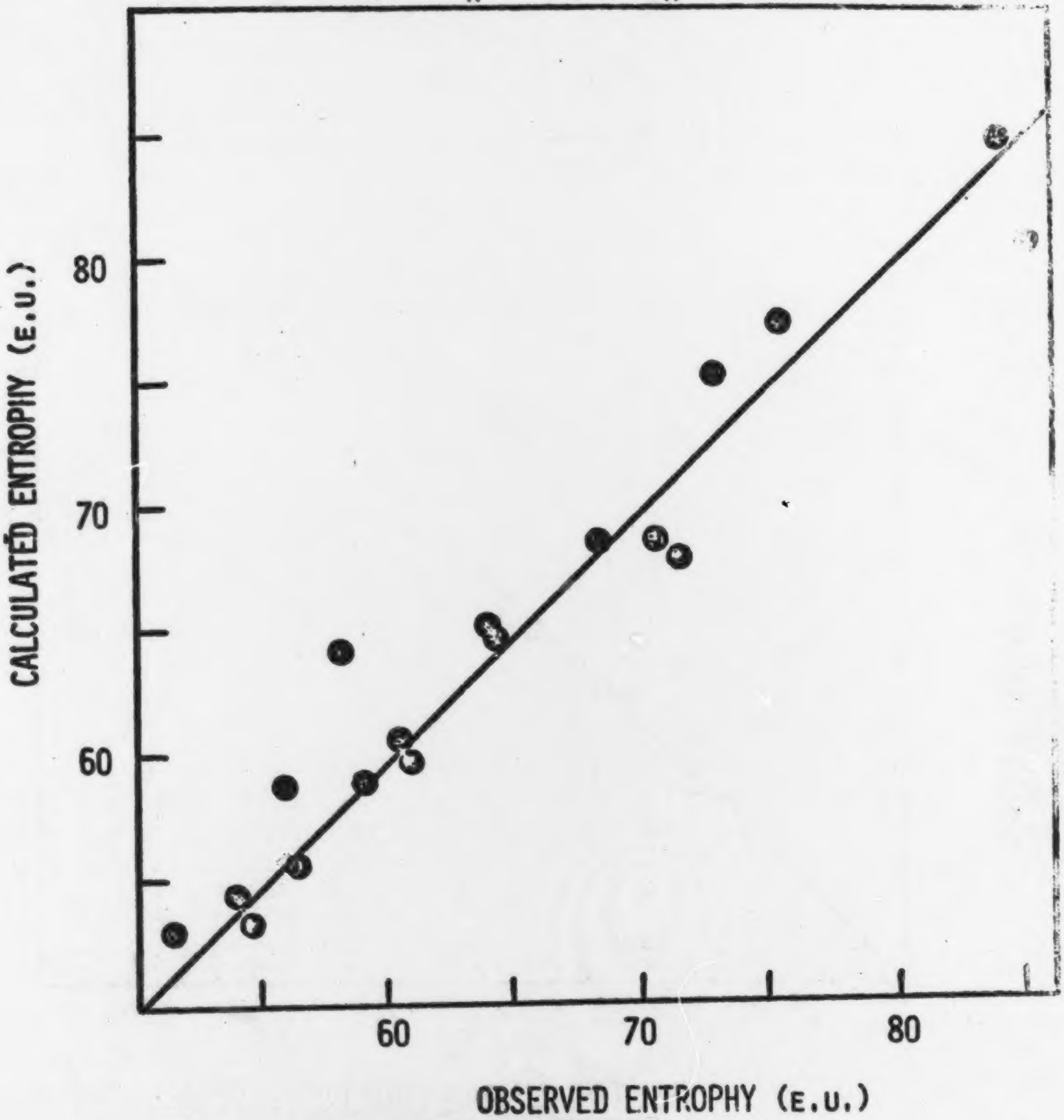
ENTROPHY CORRELATION

$S(M(OH)_N) = F(S(MCL_N))$



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expect effect for oth fp

La - med
Ce - no
Zr - no

UO2 ~~med~~ yes
iron ~~med~~ yes
chromium yes

Ru - yes but it doesn't matter

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~~C-55~~

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 1 1 1982

MEMORANDUM FOR: W. Pasedag, Section Leader
Accident Evaluation Branch

FROM: R. Sherry
Fuel Behavior Branch

SUBJECT: REVIEW OF ORNL DRAFT INTERIM SOURCE TERM REPORT

References:
1) Albrecht, H. and Wild, H., "Investigation of Fission Product Release by Annealing and Melting of LWR Fuel Pins in Air and Steam," Paper presented at the Topical Meeting on Reactor Safety Aspects of Fuel Behavior, August 2-6, 1981, Sun Valley, Idaho.
2) SAND74-0382, "Core Melt Experimental Review," March 1977
3) NUREG/CR-1410, "Report of the Zion/Indian Point Study," Volume 1, August 1980
4) NUREG-0850, "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating their Effects," Volume 1, November 1981

I have completed my review of the ORNL report. I think the results of this study are interesting from the standpoint that independent scientists using the same data base can obtain widely different estimates for accident source terms. This study also shows that it is not intuitively obvious that source terms have been grossly overestimated by past-studies as has been claimed by some. What these studies do show (and what I firmly believe) is that source term predictions have large uncertainties. I believe that the ORNL predictions are within the range of uncertainty (as are the EPRI predictions). However, I don't believe that the quantitative ORNL results are the best estimate values.

I believe that the ORNL source term estimates are high. The principal reasons they are overestimates are:

1) Fission product release during the phase of the accident when the molten core is attacking the lower reactor vessel are being over predicted. The reasons these releases are being overestimated are that release rates from small scale tests (approximately 100g) are being used for very large volumes (approximately 10-100ton) of materials. The widely different S/V ratios and the enhanced importance of mass transport resistance within the melt is being neglected.

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Note that Albrecht calculates much lower releases for this phase of the accident based on his SASCHA data and on an assumed temperature of 2400°C (Ref. 1).

Another reason that the releases are high may be because the melt temperatures during this phase of the accident are being overpredicted. I don't know what temperatures were calculated, however. (I think this type of information should be included in an appendix to the report.)

2) Fission product release during the melt/concrete interaction phase of the accident are higher than what I would expect. Melt temperatures should be, in general, much lower than for the previous in-vessel phase and this should limit the releases. The effective surface area of the melt is much larger, however, because of the large volumes of gas which are passing through the melt. The gas flow and the more chemically reactive environment could significantly enhance release on the other hand.

The values contained in the ORNL report are based on the information presented by Haaland (Ref. 2). The values given on Table B.4 (1st column) are the range(s) estimated by Haaland. The lower value being that calculated by Parsley and Fontana (for the Reactor Safety Study) and the upper value being calculated by Haaland assuming a factor of 10 greater gas release from melt/concrete interactions than Parsley and Fontana assumed. The fission product release value assumed as a best estimate by ORNL is typically within the range (between Parsley and Fontana, and Haaland), but is much closer to the higher (Haaland) estimates. I believe these (ORNL) estimates are much to high. First of all, the total gas generation value assumed by Parsley and Fontana ($1.25(10)^6 ft^3$) agrees quite closely with recent estimates made using the advanced (WECHSL) core/melt concrete code ($1.2(10)^6 ft^3$) (Ref. 3). Secondly, the temperature (3000°K) assumed by Parsley and Fontana is much higher than the peak molten core temperature estimates I have seen for this phase of the accident. For example, MARCH calculations performed for NUREG-0850 (Ref. 4) predict initial melt temperature at the time of vessel failure of from 2150°C to 2450°C.

Since fission product (and aerosol) release rates are a very strong function of temperature, I would expect even the estimates of Parsley and Fontana to be high.

3) I don't know what the bases was for the leach estimates given on Tables B.13, B.14, and B.15. It is claimed that these values are the fractional losses 10 days after water contacts the failed fuel rods. These numbers, however, are very much larger than Katayama's leach data quoted in Table B.2. Whats going on - did someone forget that the values in Table B.2 are a percentage and not a fractional value as used in Tables B.13, B.14, and B.15? It certainly looks that way.

170 1

~~C-57~~

MAR 11 1982

In addition to the above comments, I believe the importance of primary system retention has been underestimated by ORNL as well as the effects of delayed containment failure on source term reductions. Enclosed are my detailed comments on the report.

R. R. Sherry
Fuel Behavior Branch
Division of Accident Evaluation

Enclosure: As stated

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Detailed Comments

P-10, Last paragraph, 4th line - "chemical and physical processes and engineered safety features".

P-11, last paragraph, 7th line - Volatilization? (vaporization? i.e., core melt/concrete release).

P-14, Section 3.4.1.3 - Where are the total energy release rates given - did I miss them some where in document?

P-17, Table 3.1 - A companion table should be shown which shows the research currently underway to improve the technical bases.

P-17, Table 3.1 - "Transport through containment" - "first principles code." I believe most required data is available. In fact, most of the mechanisms are included in one or more of the existing containment codes.

Appendix A

P A-2, 3rd sentence from bottom - "entirely dry environment" must be defined (e.g., superheated steam and hydrogen environment).

P A-4, first paragraph, 7th sentence - I don't believe that a majority (or even a large minority) of the fuel rods will fail for a large LOCA (with successful ECCS operation) if one looks at it realistically.

P A-5, 5th sentence from bottom - "bypass the drywell (not containment)".

P A-7, 9th sentence - "However, if water were present in the pressurizer or pressurizer quench tank".

P A-7, 3rd sentence from bottom - "the containment and containment safety features".

Appendix B

P B-2, Item 2) - "Diffusion from fuel matrix prior to melting".

P B-3, 2nd paragraph - Rapid release of volatile radionuclides occurs at the time of cladding rupture and then more slowly as volatile species evolve from their condensed phase (to try to reestablish equilibrium, e.g., condensed phases of cesium uranate and cesium iodide release gas phase constitutes Cs and CsI, respectively) which are subsequently released. (I don't buy this additional release mechanisms of shallowly imbedded gas.)

P B-3, 2nd line from bottom - Delete "at temperatures above the melting point of UO$_2$" (approximately 2800°C) and insert just "from molten fuel". The temperature at which the UO$_2$ fuel becomes liquid in a fuel rod is predicted to be much less then stoichiometric UO$_2$ melting point because of mixing with lower melting point components (e.g., cladding).

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P B-5, 8th line - (mainly H_2, CO, CO_2, and steam).

P B-5, last line - "thick crusts of concrete decomposition materials (slags) over".

P B-6, 2nd paragraph - same as previous comment on shallowly imbedded gas.

P B-6, 2nd paragraph, 6th line - Replace "plenum" with "Interconnected voidage (i.e., plenum, fuel cladding gap, cracks, pores, etc.)".

P B-7, line 6 - Replace "do" with "did".

P B-8, 5th line from bottom - Rewrite sentence, biggest effect of oxidation (steam explosion) release was the predicted increase in the Ruthenium group release fraction (due to formation of volatile oxides). The correct values are shown on Table B.1.

P B-12, 4th sentence - What is "at best" here supposed to mean (imply).

P B-18 - Note, Leach values on this table are a percentage. Later on Table B.13, B.14, and B.15 these values seem to be interpreted as fractions.

P B-20, 2nd paragraph - Note that FASTGRASS models only the pre-melt heatup phase of the release.

P B-21, 2nd paragraph - Iodine chemistry has been considered important for a long time.

P B-21, 2nd paragraph - Rewrite last sentence - garbled.

P B-30, first paragraph - Preliminary information on control rod material release rates are available from Albrecht (KfK) and from Parker (ORNL). See Reference 1 from cover letter.

P B-31, Section B.5.2, line 3 - "molten core materials, (UO_2, control rods, cladding, and structural materials).

P B-31, last two sentences - Using the empirical release rates derived from small scale experiments for the releases from the molten fuel during the phase of the accident when the fuel is in the lower reactor plenum is probable excessively conservative because of large differences in the surface/volume ratios. The melt release model in the START (BCL) computer code could probably be used to generate better estimates than directly using the (ORNL) empirical release rate correlations. During this phase of the accident, it is also very important to specify correctly the debris temperature (and time at temperature). In this regard the temperature histories of the debris should be supplied for a few representative sequences.

P B-32, line 8 - The INTER code is largely obsolete. CORCON-MOD-1 and WECHSL are much more advanced codes and are available for analyzing melt/concrete interactions.

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P B-35, 11th line from bottom - "WECHSL or CORCON which are an improvement".

P B-36, first line - Aerosols are mainly concrete decomposition products.

P B-36, 2nd paragraph, 2nd line - Add CORCON.

P B-38, 2nd paragraph, 7th line (also P B-39, line 1) - Replace "containment building" with "reactor cavity".

P B-40, line 14 - Replace "melting temperature of the UO_2" with "temperature at which the fuel rods are assumed to melt." Reason - mp of UO_2 approximately 2800°C, mp of fuel rods (cladding + UO_2) may be as low as approximately 1850°C (mp of cladding).

P B-40, paragraph 2, line 4 - What has reduced 50% of the UO_2 to U? Where did this assumption come from?

P B-42, Section B.6.3, lines 7 and 8 - Replace "cladding melting" with "cladding failure".

P B-42, Section B.6.3, line 9 - Replace "fuel melting" with "incipient melting of the fuel rods".

P B-53 - Note previous comment on Leach values for Tables B.13, B.14, and B.15.

P B-53 - What is the peak temperature assumed for the AD "gap release." If temperatures were assumed not to exceed 1200°C, (I doubt they would) then the escape fractions (volatilization) are excessively high. The ORNL gap release model (Lorenz, Malinauskas, et.al.) was obviously not used for the Cs and I calculations for this table - why not?

P B-55, Table B.16 - Aerosol releases prior to RV failure seem low. For example, 280 kgs for event V is made of approximately:

140 kg	Cs
12 kg	I
23 kg	Te
30 kg	Mo
35 kg	Ba
10 kg	Sr
8 kg	Zr (FP)
2 kg	Ru
260 kg	

Indicating only a minimal amount of structural materials in the aerosol. This result is in variance to other studies, e.g., Albrecht, et.al. (Ref. 1), where it calculated that 3500 kgs of aerosols are released prior to RV failure (including 1800 kg of control rod silver, 450 kg of structural steel, and 450 kg of UO_2). Conversely, some of the estimates for aerosol release during melt concrete interactions seem incredibly high (9100 kg, 23000 kg).

P B-59, 2nd paragraph, line 9 - The fact that neither WECHSL nor INTER currently account for gas evolution from radiatively heated concrete above the surface of the molten pool is irrelevant since this gas would not contribute to sparging of fission products.

P B-59, paragraph 2, line 12 - The CORCON code has a relatively sophisticated model for chemical reactions between the gas passing through the melt and the constituents of the melt. But I don't see how the chemical interactions would effect the total volume of gas released since the typical reactions are:

$H_2Og + M \qquad MO_x + H_2$

$CO_2 + M \qquad MO_x + CO$

(M = metal species in melt)

Appendix C

P C-5, Table C-1 - Indicates chemical reactions are probably not important - I disagree. Indications are chemical reactions could be very important - in particular reactions of species such as I, Cs, and Te with structural materials or control materials (e.g., Ag) resulting in less (or more) volatile compounds.

P C-9, first paragraph, line 2 - This sentence is way to strong. Quick models, in some detail, aerosol processes in a one node region. However, in no way can it be considered to be better suited than TRAP for RCS transport calculations. For example, QUICK does not consider FP vapor forms or interaction between vapors and aerosols nor does QUICK consider multi-node geometries connected by convective flows (i.e., the RCS). Also, TRAP-MELT now has models for gravitational agglomeration and settling.

P C-11, first paragraph, last sentence - Figure C-1 has nothing really to do with RPV failure times. All this figure is intended to show is the importance of initial mass concentration and (residence) time on aerosol retention.

P C-11, 2nd paragraph, first line - Figure C-1 says nothing about I_2 behavior.

P C-11, 2nd paragraph, 5th line - "one" should be "none".

P C-14, 2nd paragraph, line 4 - To condense or chemisorb".

P C-14, 2nd paragraph, line 5 - This also depends on the relative surface areas of the respective surfaces (aerosol vs. structure).

P C-15 and C-16 - RE Maximum aerosol concentrations - Table C.3 does not appear to give an accurate portrayal of what the maximum (initial aerosol) mass concentrations would be. It assumes, I believe, simply X kg of aerosols distributed within some limited volume (subset) of the RCS. A more appropriate method for determining the maximum initial concentration is to divide the mass release rate M by the volumetric flow rate of steam and hydrogen (through the core). Hence,

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$$Ci_{max} = \frac{\dot{M}}{\dot{S}} \ \frac{(kg/sec)}{(m^3/sec)}$$

P C-19, Section C.4.1.4, first paragraph, last line - "unheated water"? How about subcooled or nonboiling as a better word than unheated.

P C-20, last several lines - The assumption that no vapor species retention occurs is probably invalid. First, chemisorbtion conceivably is a major retention mechanism for many volatile species (e.g., I_2, Te). Second, vapors that condense or chemisorb onto particles may deposit with the particles in cooler locations in the RCS.

P C-23, Table C.5 - I have major problems with these quantitative results. Where do the numbers come from? Are they guestimates? For the group A estimates the "best estimate" are always chosen as the maximum value in the range - these are strange distributions.

P C-23 - A lot of statements are made in this section which need to be supported. Why can't the current models (i.e., TRAP-MELT) be used in a predictive mode? Why isn't it possible to quantify the uncertainties to any useful degree of resolution? I don't agree with these overly sweeping generalizations. I do agree that large uncertanities exist in this area however.

Table C.6 - TRAP-MELT does include models for aerosol agglomeration and gravitational settling.

Appendix D

P D-1, 5th line - Replace "reactor" with "plant".

P D-2, last paragraph, 2nd line - Replace "reduction" with "retention".

P D-3, 2nd paragraph, 1st line - Replace "gases" with "vapors".

P D-3 - Delete "larger". All mechanisms act on particles of all sizes (mechanisms are just more effective for certain particle sizes).

P D-4, last paragraph, D-5, 3rd paragraph, D-6, last paragraph - Several places it is stated that condensation enhances agglomeration. This may or may not be the case for a given situation. Condensation increases the mass of an aerosol and hence increases its settling velocity. Condensation may also cause "long chain-like-agglomerated" aerosols to contract into spherical aerosols (which should also increase their settling velocity). Increases in settling velocity tend to increase the effects of gravitational agglomeration. However, the shape change may significantly reduce the cross section for aerosol interaction and act to reduce agglomeration rates.

P D-6, 2nd paragraph, line 2 - Delete "escaping from the containment".

P D-10, 2nd paragraph, line 4 - MULTI-AEROS (A module in the CONTAIN code) being developed by Sandia is claimed to have a model for steam condensation on aerosols.

P D-13, last paragraph - However, the amount of materials that escape from the containment at the time containment fails is strongly dependent on the predicted concentration at that time and not on the initial concentration.

P D-16, last paragraph - This writeup is confusing. What the figure shows is that up to a given concentration (correspondong to 700 kgs) there is an approximately linear relationship between initial aerosol mass concentration (source strength) and leaked mass (note that aerosol removal processes are still acting and may be significant). Above 700 kgs, the effect departs from linearity as increased aerosol concentrations promote more rapid aerosol deposition.

P D-18, MARCH III BWRs are nearly as large as some of the "large" PWRs.

P D-21, first paragraph - BWR MARCH I and II's drywell and wetwell = containment (suppression pool is in wetwell).

BWR MARCH III's drywell is inside containment - suppression pool is also inside containment. See attached figure.

P D-21, 2nd paragraph - See previous comment on effect of steam condensation on aerosol behavior.

P D-22, 2nd paragraph - Similar comment as before on MULTI-AEROS.

P D-23, first paragraph - Equal to or less than design leak rate occurs for design pressure. For pressures less than design, much less than design leak rate, for pressures in excess of design - maybe greater than design leak rate.

P D-25, Section D.4.2.4.1, first paragraph, first line - Why? Because of temperature? I can visualize situations where H_2 burns occur or where containment sprays with strongly caustic additives are operating. I think this would qualify as something more than a "region of comparatively low chemical activity." Next sentence - with exception of CH_3I or species which enter the liquid phase and hydrolyze (e.g., I_2 HOI and I^- and IO_3^-, or dissociate CsI Cs^+ and I^-, etc).

P D-26, 3rd paragraph, 5th line - Replace "reactor" with plant".

P D-27, 2nd line - The section on 0772 on organic iodine formation mechanisms should be credited to Campbell.

Table D-1 - I disagree with the comments on the importance of chemical reactions. Particularly, when I consider H_2 burns, which could have a major effect on the chemical forms of FPs suspended in the containment atmosphere or deposited on surfaces.

Table D-3 -	PWR	Ice Condenser
Main steam isolation valve ...	N/A (instead of No)	N/A

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Table D-3 - Footnote b (see attached figure courtesy of GE).

Table D-5 - Footnote d. Not really true. Shape factors are allowed (input) which account for nonspherical particle properties. In any event the "assumption" of particle shape had nothing to do with steam condensation effects.

Table D-6 - Check your values for containment free volume. Large dry PWRs are somewhat bigger. For example, Zion has a volume of approximately $2.7 (10)^6 ft^3$ [$7.6 (10)^4 m^3$] and IP has a volume of approximately $2.6 (10)^6 ft^3$ [$7.4 (10)^4 m^3$].

Table D-7 - I don't understand what the fractional values indicate for event V. Footnote b indicates they are the fraction released into containment that escapes from the containment. But for event V nothing is released to the containment.

Table D-8 - Footnote b. These are not numbers. What do you multiply?

Table D-9 - "Meltdown column" replace "reactor" with "plant" (or "containment"). (First and second paragraph.)

Table E.5 and E.6 - Are confusing and hard to understand.

Section E2, first page, first paragraph - Whats the difference between "lack of understanding" or "lack of knowledge".

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MARK III

Note - drywell pressure boundary is not part of primary containment outer pressure boundary as is the case for MARK I's & II's

MARK III CONTAINMENT

REACTOR BUILDING

1. Shield Building
2. Free-Standing Steel Containment
3. Polar Crane
4. Refueling Platform
5. Upper Pool
6. Reactor Water Cleanup
7. Reactor Vessel
8. Steam Line
9. Shield Wall
10. Feedwater Line
11. Drywell
12. Recirculation Loop
13. Weir Wall
14. Horizontal Vent
15. Suppression Pool

AUXILIARY BUILDING

16. Steam Line Tunnel
17. Motor Control Centers
18. RHR System

FUEL BUILDING

19. Fuel Transfer Bridge
20. Fuel Transfer Tube
21. Cask Handling Crane
22. Fuel Storage Pool
23. New Fuel Vault
24. Cask Loading Pool
25. Spent Fuel Shipping Cask
26. Fuel Cask Skid

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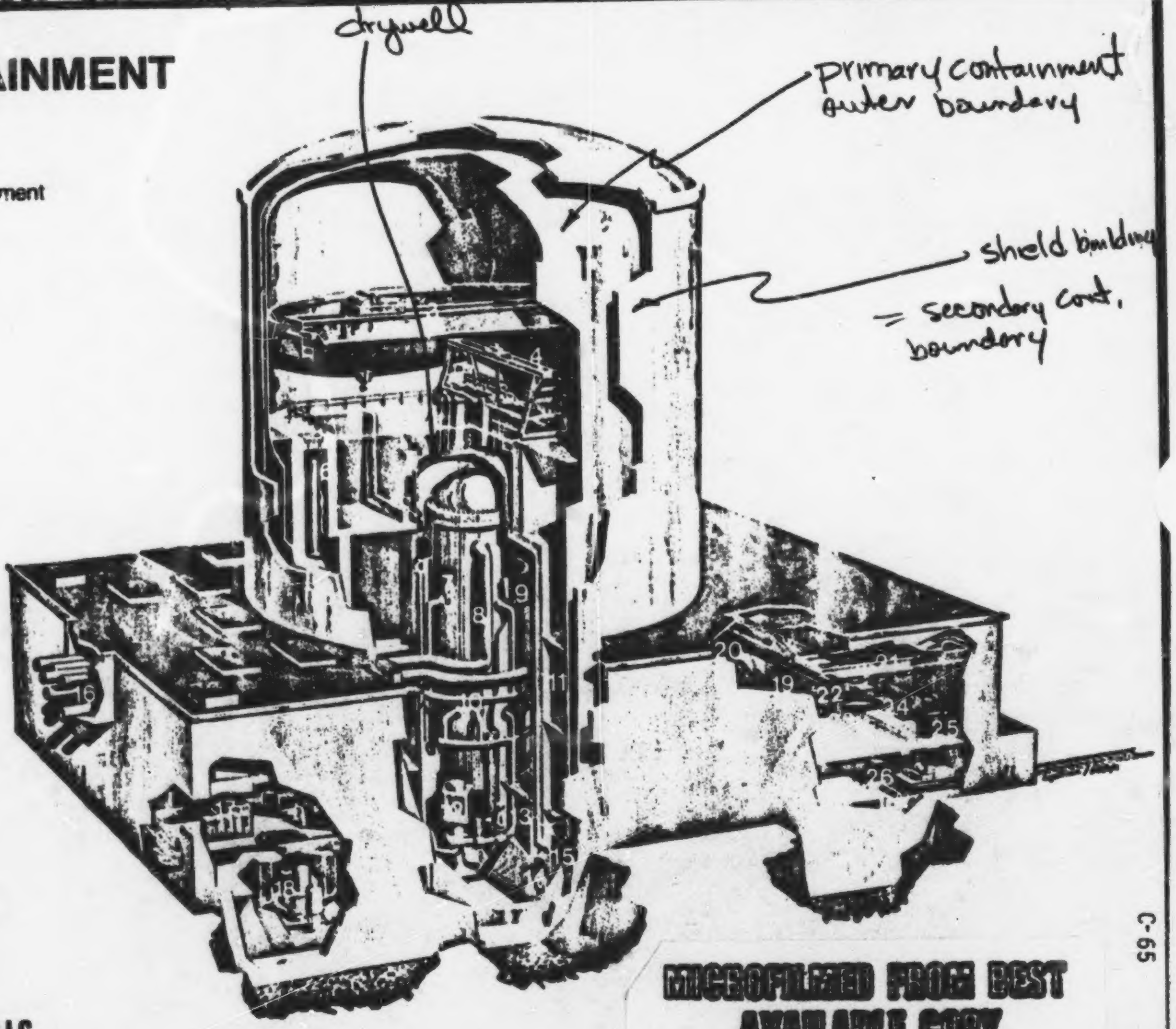
drywell
primary containment outer boundary
shield building
= secondary cont. boundary
4
6
8
9
11
13
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3
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SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-2606

MFN- 054-82

April 22, 1982

U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation

Gentlemen:

SUBJECT: GENERAL ELECTRIC COMMENTS ON ORNL/TM-8274 AND ORNL/TM-8275

On March 18, 1982 at a meeting in Bethesda, Maryland, the NRC solicited comments on ORNL/TM-8274 and ORNL/TM-8275. Unfortunately, General Electric did not receive copies of these documents prior to the meeting and was therefore not able to participate at that time. GE has subsequently obtained draft copies of these two ORNL documents. We have reviewed them and herein supply (see attached) comments on these documents.

In attempting to summarize our concerns with these draft documents we can group our comments into the following areas:

1. The authors appear to be unaware of the General Electric comments on NUREG-0771 and NUREG-0772 as these comments do not seem to be reflected in these ORNL documents. To reiterate our positions we are including our comments on these NUREGs with this letter.

2. Neither of these documents indicate an understanding of the multi-compartmentalized containment design incorporated in the BWR pressure suppression containment design. GE believes that this design significantly impedes fission product transport from a core disruptive event to the site perimeter and, therefore, needs to be considered in these documents.

3. GE emphatically endorses the concept of source term evaluation voiced in the study. However, based on the historical difficulties we have experienced with resolving the technical deficiencies in the TID-14844 values used in Regulatory Guides 1.3 and 1.7, we strongly urge that before these studies are finalized that great care be taken to resolve all peer review comments. By this we do not wish to unduly delay the adoption of these source terms; in fact we would like to expedite their incorporation into the licensing area. But

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at the same time given a choice between the current TID values and new terms which are based on a defective model and which do not represent the BWR design, we see more benefit in using the TID values.

Should you have any questions about our comments on these ORNL documents, please contact Mr. Noel Shirley (408 925-1192) of my staff at your earliest convenience.

GE would be pleased to participate further in the development of these documents.

Very truly yours,

S. J. Stark for J. F. Quirk

J. F. Quirk, Manager
BWR Systems Licensing
Nuclear Safety & Licensing Operation

JFQ:pab:sem/B04216

Attachment

cc: L. S. Gifford (Bethesda)
W. Pasedag (NRC)

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GENERAL ELECTRIC COMMENTS ON ORNL/TM-8274 AND ORNL/TM-8275

A. ORNL/TM-8274

1. Throughout this document the authors both appear to confuse PWR designs with BWR designs and show a lack of specificity in their terminology. For instance, in Section 1.1.2, the phrase "inside the reactor" is used. Does this mean "inside the reactor vessel," "inside the containment(s)," or something else? Other examples are continued references to "the" containment without a specific reference to a BWR wetwell, drywell, reactor building, etc.. Also, references made to "containments" do not reference the BWR suppression pool concept, and references are made to "hot leg breaks," which are not applicable to BWRs.

2. In Section 1.1.2, the statement is made that "current considerations for emergency planning are often based on a risk assessment approach." General Electric knows of no such philosophy or consideration. The EPZs are based on TID source term assumptions. If other considerations do exist, they should be provided in this report.

3. On Page 6, Section 1.1.3.1, it should be emphasized that NRC management has stated before the NRC Commissioners that TID-14844 source terms are not applicable for use in the environmental qualification area. This same Section, on Page 3, states that environmental qualification is based on the concept of defense-in-depth. GE believes environmental qualification is more properly defined as a defense against common mode failure.

4. On Page 7, the last paragraph in Section 1.1.3.1.2, GE believes, misses the point. The issue is not what fraction of the release comes from the entire core, but rather what fraction of the release comes from what fraction of the core. This concept is reinforced in Regulatory Position C.4.c(2) of the proposed Revision 1 to R.G. 1.89 wherein the inventory from only damaged rods need be considered when determining source terms.

5. In Section 1.1.3.1.3, the concept of the design basis is referenced. There is no design basis for which the ECCS initially fails to operate, thereby resulting in some fuel damage, and then operates to terminate the event. Such a scenario is illogical.

6. In Section 1.1.3.2.1, General Electric would like to know who the "some" are that use a "modified design basis" source term. Should such a modified design basis be relevant to this discussion, it should be provided.

7. On Page 9, Section 1.1.3.2.1, what is meant by "some considerations of emergency planning?" Greater detail is required here

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to make this a meaningful statement since GE knows of no such recent considerations.

8. The last paragraph on Page 10 needs to be more specific in acknowledging and considering the effects of the BWR suppression pool and containment compartmentalization.

9. In Section 1.1.4, the study inadequately considers the BWR pressure suppression concept and the impact this concept has on mitigating nuclide transport.

10. On Page 16, Section 1.2.1.1.2, greater consideration needs to be given to the benefit derived from BWR compartmentalization for determining containment environments.

11. Section 2.1 defines a "transient" in such a manner that normal deviation from steady state operation such as a scram, which is certainly a transient, would not fit the definition. The definition provided is couched in terms of an accident condition and should be so stated.

12. In Section 2.1, it must be acknowledged that, in order for a core melt event to occur, all ESFs must malfunction.

13. On Page 24, Section 2.1, General Electric's current ECCS model calculates no fuel perforations for a LOCA in which the ECCS functions. For such events, the source term used in this document needs to reflect this.

14. On Page 25, Section 2.2.1, the conclusion is reached that uncovering part of the core would lead to core melting. For the BWR, at least 1/2 of the core could become uncovered, and, due to steam cooling, overheating would be prevented. This fact must be recognized.

15. On Page 26, the concept of essentially instantaneous releases from perforated fuel rods appears unduly conservative. Actual release rates would be functions of both the differential pressure between the system operating pressures and those inside the fuel rod and the friction factor associated with the release rate through the perforated rod.

16. Section 2.1.2 on Page 28 implies that a steam explosion was a distinct possibility. GE has analyzed such an event and has concluded that there is no such possibility for such an occurrence in a BWR.

17. Table 2.3 contends that an effect of fission products on the Standby Gas Treatment System would be to clog the system. General Electric believes that due to many effects, such as the suppression pool, the compartmentalization of the BWR, the physical distance between the core and the SGTS, etc., clogging simply could not occur.

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4/21/82

18. Footnote b on Table 2.4 contains errors. First, "secondary" should be "primary"; and secondly, the BWR drywell can withstand much higher pressure than its design pressure at 15-25 psig. Structural evaluations have shown its actual pressure capability to be several times its design pressure.

19. On Page 43, Section 2.6, there is no cutoff provided to the range of releases. Rather than leave this open-ended, a cutoff point needs to be provided.

20. The reference to the RSS in Footnote a of Table 2.5 is not applicable to the BWR/6 Mark III containment.

21. The first paragraph in Section 6.1.1 basically implies that nothing new has occurred. This ORNL study should result in these previously untreated topics being not only addressed but also resolved.

22. On Page A-2, Section A-2, reference is made to an "entirely dry" environment. For the BWR event, there is no such environment.

23. On Page A-3, note should be made that for an event in which the ECCS performs its designed function, there are for the current GE product line fuel no calculated perforations and hence no fission gas release.

24. Section A.2 on Page A-5 states that "release nuclides would typically bypass the drywell." Except for breaks inside the drywell for the BWR, this is not "typically" but "always" and "on purpose" since the drywell is used to route fission products to the pool.

25. Section A.3.2 on Page A-8 should be specified as applicable to the Mark I containment design. Also, in the same section on the same page, the phrase "if all control rods..." should more representatively read "if all 170 individual control rods...."

26. On Page A-13, the last sentence in each of the first three paragraphs of Section A.3.4 is simply wrong. In order for the containment to fail due to overpressurization, a core melt would have to be initiated first. Containment failure would not be catastrophic but is calculated rather to be a crack, not at the base mat but rather at the transition of the top of the primary containment steel vessel reactor building wall and the roof of the building. Thus, the suppression pool is still functioning, and essentially no release is expected to occur.

27. The footnote on Page A-13 is applicable not to the BWR but rather the RSS PWR.

B. ORNL/TM-8275

1. Page 1 and Page 2 of Section 4.2 appear to provide contradictory statements about core melt releases. The footnote on Page 1

says that after a 50% melt, melting would not be stopped, yet on Page 2, a 50% melt is assumed. If the assumption on Page 1 is correct, how does the assumed melt on Page 2 stop at 50%?

2. On Page 2, Section 4.2, two statements are made about dry pathways. For the BWR, during a LOCA the pathways would be anything but dry.

3. The last paragraph of Section 4.3 on Page 4 should note that leaked fission products would, for the BWR, be in water and would stay there.

4. This document suffers from the same problem that ORNL/TM-8274 has with respect to broad generalizations about "the containment." For the BWR, any study of this sort must differentiate, since each is significant, between the applicable portions of the compartmentalized BWR containment system. This concept also applies to the broad generalizations about "reactor" when, in fact, what is meant is either "plant" or "site."

5. Section 4.4 on Page 5 notes the importance of water in the consideration of source term considerations. The section should note that the Mark III containment contains greater than a million gallons of water.

6. Table 4.2 perpetuates one of the problems with the TID-14844 source terms. This table has a 100% release, apparently simultaneously, for both airborne and waterborne sources. Table 4.4 has the same problem.

7. The sentence beginning on the last line of Page 6 is in total opposition to the statements made by NRC management to the NRC Commissioners on October 21, 1981, on the same subject.

8. GE believes the 30% uncertainty factor referenced on Page 7, Section 4.6, is unduly conservative.

9. On Page 2 of Section 5.2, the statement about "the pathway to the containment is short and unobstructed" is simply not true for the BWR.

10. On Page 7, Section 5.6, the need to assume a hydrogen burn appears to be a new regulatory position. Is it? It is not an environment to be considered for environmental qualification purposes.

11. Tables 5.6 and 5.7 appear to ignore the BWR suppression pool.

12. Page 11, Section 5.6, talks about "relative probabilities." Is this a never-ending series, or is there a probability cutoff beyond which there is no need to consider further events?

13. This study would be enhanced if sample calculations were provided showing how calculated values have been obtained.

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4/21/82

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MFN-044-81
RHB-125-81

March 17, 1981

M. Silberberg, Chief
Experimental Advanced Safety Technology Research Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Silberberg:

SUBJECT: COMMENTS ON DRAFT OF NUREG-0772

This letter provides General Electric Company comments on the draft of NUREG-0772, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents" which was provided with your letter of March 6, 1981.

The report is represented as a compendium of the best technical information available for estimating the release of radioactive material during postulated severe accidents in light water reactors. General Electric considers that the draft report has helped to focus on areas of uncertainty regarding fission product behavior from reactor accidents. In general, however, the specific conclusions regarding fission product retention are based on incomplete information on accident sequences and plant design characteristics. The various sections of the report require better integration to insure consistent unambiguous support for the conclusions.

General Electric considers that the draft report does not present technically supportable conclusions based on factual inputs, but instead presents results based on specific input assumptions. The following three assumptions appear to have the greatest impact on the report results:

1) Similar behavior of iodine and iodide for severe accident conditions

2) The magnitude of the decontamination factors utilized in accident scenarios

3) Characteristics of dominant accident sequences.

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GE believes that iodine and iodide behave differently in the suppression pool under accident conditions and that the predominate iodine form (iodide) will be effectively retained in the pool water. Further, it is GE's belief that the decontamination factors assumed are overly conservative and do not reflect current data. Finally, assumptions on the transport pathways, time to containment failure and low pressure emergency cooling system performance are in error. Additional information is provided in the attachment.

The information relating to the Boiling Water Reactor (BWR) is not complete and results in misleading conclusions on the BWR capability to contain and attenuate fission product release. Specifically, the following areas have not been adequately addressed:

1) The BWR geometry was not appropriately considered in assessing potential attenuation of fission product release. The multi-compartments in the various BWR containment designs and the capability of the BWR suppression pool were either neglected or only superficially treated. The suppression pool is expected to provide scrubbing capability to capture fission products released from degraded core accidents for dominant BWR containment failure modes. The internal design of the BWR containments with their multiple barriers for fission product release (i.e. system piping, drywell, primary containment) provide significant surface area for fission product removal by natural processes.

2) The experimental data relative to pool scrubbing was not utilized. The draft report cited only a single reference for pool scrubbing data. Additional references are provided in the attachment to this letter. These references support the use of decontamination factors decades higher than those documented in the draft report.

3) The evaluation of BWR dominant accident sequences is misleading. The report subjectively assesses three categories of accidents and makes assumptions on plant conditions, containment failure modes and resulting fission product transport and release. As demonstrated by the most recent BWR probabilistic risk assessment studies, this subjective consideration does not properly consider the BWR capability to prevent severe core degradation and mitigate the consequences of such degradation.

Based on the concerns addressed herein, it is recommended that the draft report be modified to include a more representative analysis of fission product retention based on the use of realistic decontamination factors. If current available data is found to be lacking in a specific area, an assessment should be made to identify the potentially attainable decontamination factors and the requisite test programs necessary to

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define them. If the draft report is presented to the Commission in its current state, appropriate characterization of its incomplete status must be highlighted. It is GE's concern that the report misrepresents current LWR fission product retention capability.

It is General Electric's intent to provide the NRC with comments on the draft of NUREG-0772 in the peer review meetings March 17 and 18, 1981 in Washington, D.C. GE will provide additional comments prior to April 1, 1980. We would be pleased to provide further details on the information provided herein. Specific questions may be addressed to Mr. K. W. Holtzclaw (408) 925-2506 or Mr. J. M. Smith (408) 925-5722 of my staff.

Very truly yours,

RHBuchholz

R. H. Buchholz, Manager
BWR Systems Licensing

RHB:rf/7N

Attachment

Attachment to
GE Letter MFN-044-81

GENERAL ELECTRIC COMPANY COMMENTS
ON
NUREG-0772 (DRAFT)

I. GE COMMENTS ON REPORT CONCLUSIONS

CsI or HI dominant form, but some I_2 in some situations.

> GE agrees - except we do not agree that I_2 (elemental) is a likely form in any BWR accident scenario.

Iodine chemical form does not have a major influence on consequences.

> GE disagrees - even under saturated conditions the pool DF for iodide is much higher than elemental I. If proper credit had been given for the saturated pool iodide DF, this conclusion would not have been reached.

Consequences have not been overestimated by orders of magnitude.

> GE disagrees - since this conclusion is based on the assumption that the pool is only worth a factor of 1 to 10 for the worst accident scenarios. The factor of 1-10 is based on a saturated pool and elemental iodine; however, as noted in the report, I_2 is not the dominant form. Also as noted in Appendix E, even for a saturated pool and for elemental iodine, the DF is at least 100 for BWR accident conditions.

Additional Comments

> The report in a number of areas indicates a general lack of understanding of the BWR barriers, geometries, and compartmentation. Everything is in terms of release from fuel or RPV directly to "the containment", as if all containments were open, dry containment designs. No understanding is apparent that Mark III has extra barrier (drywell and pool) within primary containment.

II. GE COMMENTS ON REPORT ASSUMPTIONS

The above conclusions are based on the following assumptions which are not supported by the data given in the report draft:

Behavior of iodine and iodide in the suppression pool is similar under severe accident conditions.

> Iodine and iodide behave differently in saturated pools. Partition coefficients for Iodine are $\geq$ 100 and for iodide are orders of magnitude greater, approaching infinity for some cases (see report p. 5.26 and E.4). This means that essentially all iodide will be scrubbed from small noncondensing bubbles for the calculated bubble rise times.

Assumed decontamination factors (DFs):

The DF's assumed for the saturated pool accident scenarios are far too conservative and do not reflect current data. The values used for the analysis are 1-10 whereas information presented in Chapter 5 and Appendix E support DF's greater than 100; other data (see attached references) support DF's as high as 10^4.

Modeling and assumptions used for dominant severe accident sequences:

The transport pathways were not adequately descriptive of the differences in the large dry containments and Mark I and Mark III designs. For example, sequences ended with transport to the drywall without mention of transport through the suppression pool (p. A. 13). The time to containment failure in Table 8.2 is too short for the Mark I AE sequence, it should be ~ 5 hours not .81 hours. Also, in the Mark I TW sequence the low pressure core cooling systems (LPCI/LPCS) for newer BWR/4 designs and all BWR/5 and BWR/6 designs will not cavitate under saturated conditions.

III. GE COMMENTS ON IMPACT ON RISK

General Electric calculates that the best estimate attenuation factors for the severe accident sequence show factors which are orders of magnitude greater than the 2-10 stated in the report conclusions. Using the proper DF's, the consequences of these accidents will be decades less than those presented in WASH-1400. For many sites, no acute fatalities are expected for these sequences.

RHB:rf/736-7, 70
3/17/81

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POOL SCRUBBING TEST REFERENCES

1. "Fission Product Entrainment Evaluation Tests for the Pressure Suppression System," GEAP-3206 (1959), General Electric Company.

2. McGoff, J.J., Rodgers, S.J., "Simulation of Container Venting Under Seawater," Technical Report 59, Contract NOBS-65426, Mine Safety Appliances Co., Gallery, PA (December 1957)

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RHB:rf/738
3/17/81

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~~C-78~~

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NUCLEAR POWER SYSTEMS DIVISION

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MFN 058-81

March 31, 1981

M. Silberberg, Chief
Experimental Advance Safety Technology
 Research Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Silberberg:

SUBJECT: ADDITIONAL COMMENTS ON DRAFT OF NUREG-0772

Reference: R. H. Buchholz letter to M. Silberberg, "Comments on Draft of NUREG-0772," March 17, 1981

This letter provides additional General Electric Company comments on the draft of NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents." The comments herein are intended to supplement the observations provided in the reference letter and the input provided by the GE representatives at the meeting of the Peer Review Group March 17 and 18, 1981, in Washington, DC.

As noted in the reference letter, General Electric considers the draft report to be a useful aid to focus on important areas regarding fission product behavior from reactor accidents. It is our expectation that a clear, complete, and technically accurate report addressing these important areas could provide a uniformly accepted basis for all future accident consequence evaluations, which in turn will be the cornerstone for important regulatory decisions. However, the draft report, as it is currently written, is incomplete and does not present technically supportable conclusions. GE considers that the report should not be issued or sent to the Commission without significant modifications to correct what we believe are misrepresentations of LWR fission product retention capability. In light of the importance of the report, the requisite changes should be made, and additional peer review completed, before the report can be considered to reasonably represent the state of the art.

As expressed in our initial comments in the reference letter, in our input in the Peer Review meeting, and in the attached supplementary technical comments, NUREG-0772 and its companion document on regulatory impact must properly evaluate LWR design and system capability. From our review of the draft report and participation in the Peer Review, it was evident that the Boiling Water Reactor (BWR) system and containment capabilities for fission product retention and attenuation were incorrectly treated. The following principal areas of concern are inadequately addressed by the report. Additional specific detailed comments have been provided in the attachment to this letter.

1. The analyses supporting conclusions regarding BWR accident sequences make assumptions on core coolability, plant conditions, and containment failure modes which are in conflict with the most recent BWR probabilistic risk assessment studies. Throughout the Peer Review, it was evident that there is a pressing need to clarify and examine the assumptions and analytical models employed in the evaluations of the BWR accident sequences to determine their technical adequacy.

2. The conclusion of a best estimate iodine attenuation factor for the risk dominant accidents does not consider the available pool scrubbing data presented in the attachment to the reference letter. Furthermore, it appears that unnecessarily conservative interpretations of the attenuation data have been made in the conclusion of the report that are not substantiated in the report and appendices. Using decontamination factors supportable by current technical data, the report conclusions will change, and accident consequences will be orders of magnitude lower than represented. This is in direct disagreement with a principal conclusion of the draft report.

3. As previously noted in the reference letter, the BWR transport pathways in the containment were not adequately considered. The BWR geometry and containment internal design must be reevaluated in assessing the attenuation capability. GE has recently made presentations to the NRC Staff and the ACRS illustrating the BWR release pathways and quantifying the expected significant attenuation capability.

With regard to Item 2 above, GE strongly recommends that NUREG-0772 establish realistic or best estimate values of fission product attenuation factors as opposed to conservative estimates. The realistic values should be utilized in analyses to properly characterize risk in assessing fission product transport and release resulting from postulated accidents. The risks established using realistic best estimate inputs are appropriate to assess plant performance to support the various rulemaking efforts currently under consideration by the NRC (including Degraded Core, Minimum Engineered Safety Features, Siting, etc.).

In summary, NUREG-0772 must be modified to properly consider the LWR systems and accident sequences to properly assess fission product behavior. GE is eager to initiate a technical liaison with the National Laboratory contractors involved in the accident sequence evaluations to provide consultation on BWR system capabilities to correct the deficiencies in the analytical support of the report. These meetings should be held immediately to provide substantive technical inputs for incorporation into the final version of NUREG-0772. In addition, General Electric will prepare for April submittal a detailed documented review of all pool scrubbing tests and data referred to in the attachment to the reference letter for consideration in the final version of NUREG-0772.

It is recommended that the next draft of NUREG-0772 be issued only after additional peer review. This would provide the necessary technical exchange to address all the problems of the draft report and insure a high quality final document. Any draft not addressing all areas receiving peer criticism should be clearly characterized as preliminary, with significant technical uncertainties under review to avoid distribution of misleading information. This same review process should be applied to the companion document of NUREG-0772 on regulatory impact.

General Electric would be pleased to provide further details on the information contained herein, in addition to supporting future reviews. Specific questions may be addressed to Mr. K. W. Holtzclaw (408) 925-2506 or Mr. J. M. Smith (408) 925-5110 of my staff.

Very truly yours,

S.J. Stark for R.H. Buchholz

R. H. Buchholz, Manager
BWR Systems Licensing

RHB:sem/1127-29

Attachment

GE COMMENTS ON NUREG-0772 DRAFT

The following technical comments have been developed by General Electric following a detail review of NUREG-0772 and participation in the peer review meeting. The comments have been divided into two categorgies with each comment referencing applicable pages in the draft of NUREG-0772 dated March 6, 1981. The comments presented in Category A identify specific technical concerns with NUREG-0772 and provide GE's basis for concluding that the draft document underestimates the fission product retension capability of the BWR. Category B comments identify areas of NUREG-0772 where clarification or revision of the draft text is considered necessary.

Category A

1. Pg. i - The draft concludes that the assumed form of iodine does not have a major influence on the estimated iodine release to the environment for the risk dominant accidents. This conclusion appears to be based on the results of Chapter 7 wherein it is assumed that containment failure has occurred or complete bypass of the containment has occurred, such as the TMLB' sequence. This sequence is not applicable to a BWR and the report does not identify any sequences for the MK III BWR which result in bypassing the suppression pool. The report needs to address differences in containment and reactor designs before arriving at general statements. While one could assume that containment failure resulted in failure of the suppression pool or drywell, these assumptions are only valid if a structural design analysis shows that to be the case. One could also assume that the scrubbing efficiency of the suppression pool is independent of the chemical form entering the pool. However, this assumption is not supported by the data in the report which shows iodide more effectively scrubbed than iodine.

2. Pg. ii - The statement is made that "the results of this study do not support the contention that the predicted consequences for the risk dominant accidents have been overpredicted by orders of magnitude in past studies. For example, the analysis in this report indicates that the

"best estimate" (underline added) attenuation factor for iodine is between 2 and 10 for the risk dominant accidents..." It appears that in a number of areas a conservative rather than a best estimate interpretation of the data has been chosen. For example, a partition factor of 100 was the minimum value reported compared to a possible partition factor of 10^5 (see pg. 15). A decontamination factor (DF) of 1-10 was employed for the suppression pool. Appendix E would suggest a minimum DF of 30 (for 2 micron - 4 gm/cc particles, 1 cm bubbles, 250° F saturated pool and a 15 second transient time) and a maximum value of infinity for 40 micron particles or iodine vapor entrapped in 1 cm bubbles with a rise time of 10 seconds. Based on this data the 2-10 estimate is overly conservative for the BWR sequences.

3. Pg. 16 - The statement is made that "for the most severe accidents analyzed for the BWR suppression containment" the attenuation factor applicable is 4. It would appear that this conclusion is based on the assumption of zero removal credit by the suppression pool, or bypassing the pool by either drywell failure or suppression pool failure. Relative to zero removal credit by the suppression pool, it would appear in the transcript of the Peer Review (pg. 259) that a "violently" boiling pool was assumed.

   GE has not identified any accident sequences where the pool would be "violently" boiling and result in a zero pool decontamination factor during or after the period of fission product release from the core. In those sequences where containment failure precedes core melt (and the pool is saturated), containment depressurization has already taken place. The pool is relatively "quiet" when the small bubbles of non-condensing steam rises through it and significant scrubbing of particulates and iodine is expected. In those cases where significant fission product release has occurred prior to containment failure, the pool is not in a violently boiling state and therefore provides an effective barrier to fission product migration. Therefore, even assuming violent boiling occurs after containment failure, only a small fraction of the liquid is released due to boiling, and hence only a small fraction fo the contianed activity is released.

Relative to the posulated containment/drywell failure modes, it is necessary that careful consideration be given to design features of the particular containments. For example, while one may postulate, for very low probability events, failure of the free standing shell of the Mark III containment due to overpressurization, the drywell and pool would in all probability remain intact. Since all fission products escaping the RPV would still have to be passed through the suppression pool for this failure mode, the fission product retention would remain very high.

4. Pg. 17 - The statement is made that "scrubbing of particulate iodine would be less efficient in sequences involving limited core damage than for more severe sequences. In the severe accident sequences scrubbing efficiency would be comparable or better than has been predicted for elemental iodine." It is difficult to understand how this conclusion can be reached when only 1 scrubbing test has been referenced. Scrubbing of particulates is also dependent upon the wetting and solubility characteristics of the scrubbed particle as well as the bubble size, residence time in the pool and bubble media, i.e. steam, air, CO_2, etc. The text does not provide sufficient data to allow an independent assessment of the validity of the report conclusion. It is therefore recommended that the rationale and the applicability of the conclusions be included in the report.

5. Pg. 5.25 - The calculated partition coefficients shown in Figure 5.5 are not consistent with the experimental data reported in the literature (Ref. 2 and 3 see attached figures). At least a factor of 10 - 100 larger should be expected at lower iodine concentrations. It is also important to emphasize that, as given on page 5.26, "the calculated partition coefficients in this paper have assumed an initial iodine source of molecular iodine, I_2, dissolved in water and reacted with water to produce the equilibruim concentrations of iodine species. It should be obvious that solutions where the only iodine species is iodine, I^-, will have essentially infinite partition coefficients." (See comment B6 for application of partition coefficients in pool scrubbing calculation).

6. Pg. 5.27 - On pg. 15 the statement is made that methyl iodide is judged to be less than 0.1%, however, on pages 5.27 and C.26 it is recommended that a value of 0.03% be assumed. However, this value is based on the conversion of I_2. Also on page C.26 it is noted that TMI resulted in only a few thousandths of 1% conversion. Based upon the information included in the report and recognizing that the expected chemical form of iodine is iodide not I_2 a best estimate conversion factor would be 0.005%. The report needs to arrive at a recommended best estimate value not a lower bound value. Also, consistency with the data in the total report is required as well as consideration of containment designs, accident scenarios and fission product scrubbing barriers such as the BWR suppression pool.

7. Pg. 7.23 - It would appear that the assumptions of containment failure in the region of the suppression pool or negligible retention in a boiling pool where used in arriving at the conclusions on pages i and ii. i.e. the assumed chemical form of iodine is unimportant in risk dominant accidents and the best estimate attenuation factor for iodine is between 2 and 10. As pointed out in the previous comments, the assumption of suppression pool failure is not valid for all BWR containment designs and the assumption of DF in a boiling pool is not justified for any containment design. Therefore it is GE's opinion that the chemical form of iodine is important for all accident sequences.

8. Pg. 8.6(a)- It is noted that the AE accident sequence involves failure of the primary containment due to overpressure with leakage through the annulus into the reactor building. It should be noted that prior to postulated containment failure significant fission product scrubbing by the subcooled suppression pool will occur, thus limiting the potential radiological consequences of this event.

   (b) It is noted that the TWc sequence resulted in containment failure in 54 hours and results in a "high temperature" of 592°C as indicated in Table 2. It is assumed that Table 8.2 is meant instead of Table 2; however, there is only a TW or a TC sequence identified not a TWc sequence. The time of containment failure is 1.5 hours and the "peak atm. temperature" is 592°C for the TC sequence and 45 hours and 409°C for the TW sequence.

It is not clear if these temperatures are "seen" by the drywell or wetwell, as implied on pg. 8.6, or are temperatures which exist within a very small volume of gas within the drywell. It is also not clear how these values have been used in assessing the effectiveness of ESF systems in general and the pressure suppression system in particular.

(c)- It is stated that the MK III containment does not include the annulus or the SGTS features of the MK I containment. It should be noted that essentially all MK III containments have both a primary containment and secondary containment structure. Leakage is from the primary to secondary containment with secondary containment being treated by the SGTS.

9. Pg. C.11 - Chemistry of Iodine (a) - The chemistry of iodine has been fairly well examined in terms of thermodynamics. It is reasonable to assume the kinetics of the reactions are generally very fast at higher temperatures, but in many cases, the concentration of the reaction product is determined by the availability of the reactants and the necessary conditions involved in the reaction. For example, the oxidation of I^- by O_2 (eq. C.5.11)

$$2\ I^- + 1/2\ O_2 + 2\ H^+ \longrightarrow I_2 + H_2O$$

is thermodyamically possible, but the reaction would occur only in acidic solutions and under the influence of light.

It is obvious that inside the RPV this reaction would not proceed because the water is basic due to the Cs^+ ions, and the oxygen content should be very minimal due to the presence of H_2. Therefore, ~ 20 Kg of iodine released within the RPV as I^- should stay as I^-.

(b)- The document fails to mention the important reaction of iodine species with the reactor material. With the excellent reducing nature of metals. at high temperature, Zr, and Fe in the core and steam separator/dryer should provide a powerful reducing medium for iodine. The reactions expected to occur include:

$$I_2\text{ (or I)} + Zr \longrightarrow Zr\ I_2\text{ (or Zr }I_3)$$

$$I_2\text{ (or I or HI)} + Fe \longrightarrow Fe\ I_3\ (+ H_2)$$

The products of these reactions Zr I_2, Zr I_3, or Fe I_3 are very stable, non-volatile (Ref. 1), and therefore should not be transported out of the RPV.

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CATEGORY B

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1. Pg. 4.22 - The ordinate is identified as fraction/min., however, for I, Xe and Kr the maximum value is ~9 which is ~9 times greater than can be physically achieved, i.e., maximum possible fraction is 1.0. It is not clear if this error is also included in the calculated time dependent release rates or is just an error on the graph.

2. Pg. 4.26 - It is noted that between 510 Kg (1100 lbs) and 1600 Kg (3500 lbs) of material would become airborne as a result of the postulated core/concrete interaction and that the resultant spherical particles would be 2 micron in mean aerodynamic diameter. It is difficult to understand how such a large mass release (1/2 ton - 1 1/2 tons) can result in such a small particle size. It is not clear if the initial particle size of the volatilized material is 2 microns and it remains as 2 microns, or the particles rapidly agglomerate into large particles resulting in rapid settling leaving only a small airborne mass of 2 micron particles.

   Considering a drywell volume of 5000 m^3, as noted on pg. 7.9 and based upon the above mass release, one would calculate a potential airborne concentration in excess of 100 gm/m^3. On pg. 7.10 it is noted that the QUICK code would predict particle sizes between 40 µm and 300 µm for an airborne concentration of 200 g/m^3. Further, the code predicts that within 30 secs, 1/2 of the particles are $\geq$ 100 microns and in 1 minute 98-99% are $\geq$ 100 microns. (Peer review transcript pg. 236) The use of a particle size of 2 microns in NUREG 0772 is inconsistent with this data.

3. Pg. 4.30 - The statement is made on pg. 13, 4.30 and implied on pg. B.3 that the release of noble gases, cesium and iodine from the core is essentially complete in 18 minutes. This statement can only be valid for a given core geometry, power density, fuel design, etc. The appropriate qualifying words need to be included in the document which explain the applicability of these values.

4. Pg. 7.2 - Reference is made to the model in Appendix E which mathematically treats vapor and particulate transport from a rising bubble. However, it does not appear that the results of the model calculations are included on pg. 7.16. On pg. 7.16 a pool DF between 1 and 10 is presented, however, the Appendix E model would predict a DF between 12 and infinity. Such inconsistencies should be resolved.

5. Pg. 8.14 - It is implied that if the inboard MSIV fails to close,that the MSIV leakage control system (MSIVLCS) is incapable of performing its function. It should noted that the MSIVLCS will perform its intended function whether the inboard MSIV is open or closed. If the inboard MSIV fails open or has unacceptable leakage characteristics the inboard MSIVLCS will be isolated and a suction will be taken downstream of the outboard MSIV.

6. Pg. E.4 Pool Scrubbing

   a) Equation E-9 describes the scrubbing factors (SF) for iodine vapor in a water pool. It is not clear what the relationship is between K_2, the overall mass transfer coefficient for vapor transport, and the iodine partition coefficient. Obviously it is inconsistent to suggest that in the most likely conditions the iodine species in the vapor phase would be CsI or HI, and use a minimum partition coefficient for I_2 of 100 in the calculation using Eq. E-9. The SF should be much larger if the appropriate value (~ infinity) for the partition coefficient for CsI or HI is used.

   b) It is also important to point out that a key parameter in gas bubble scrubbing is the solubility or chemical reaction of the species in water. For the pool scrubbing process for CsI particles (Pg. E.13), it is wrong to assume CsI particles are inert particles and treat them as such. CsI is a very soluble species,and either as a vapor or a particle readily dissolves in water. The model equation (E-30) may require modification if the above mentioned property of CsI is not properly considered in the model.

References

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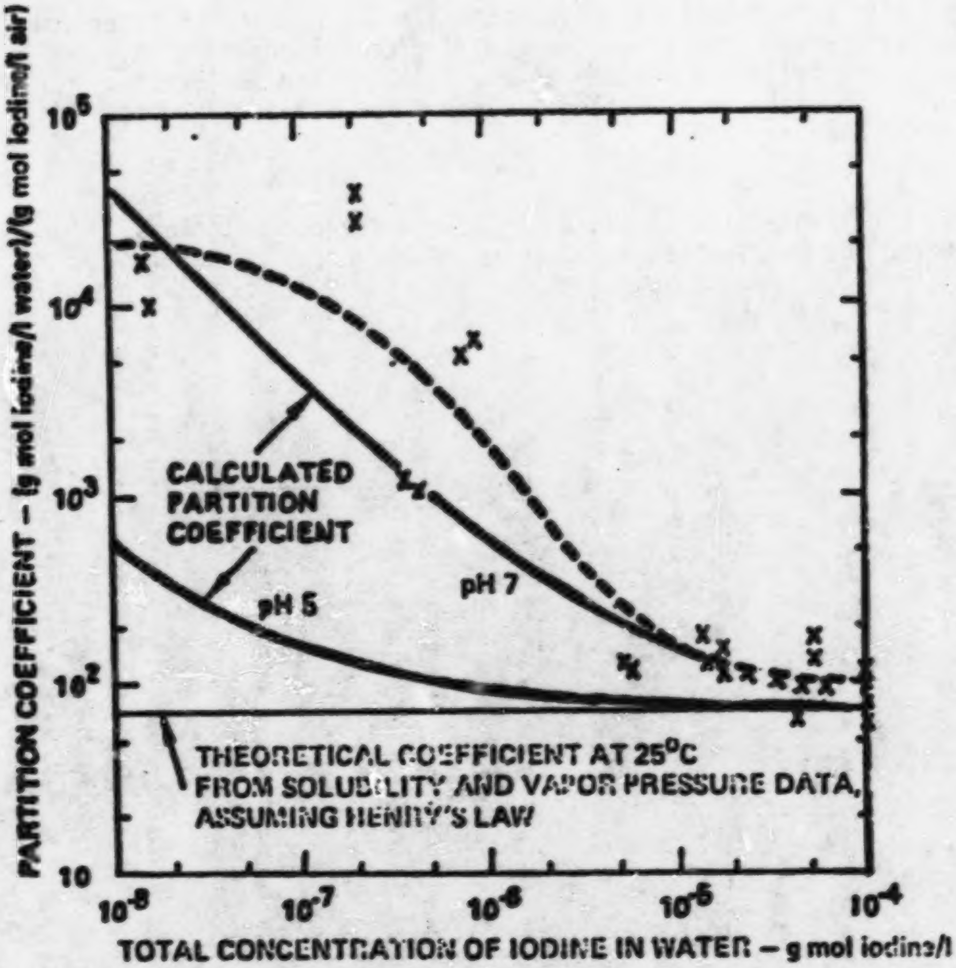

FIGURE B-2

EFFECT OF IODINE CONCENTRATION ON THE EQUILIBRIUM COEFFICIENT FOR IODINE AT 25°C

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(Ref. 2)

Docket RM-50-2
Draft Regulatory Guides
Feb. 20, 1974 U.S. AEC

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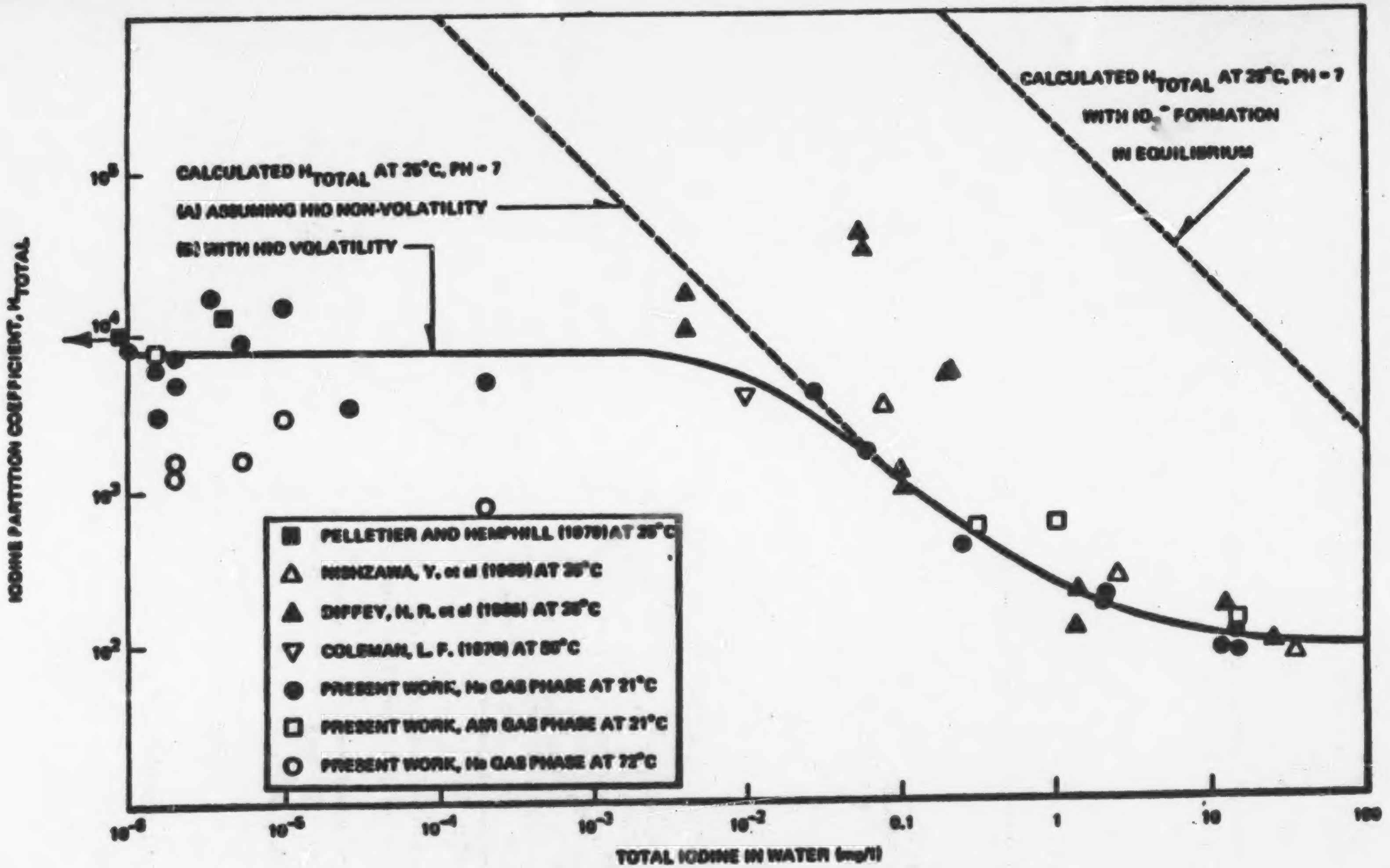

(Ref. 3) Accepted for publication in J. Inorg. Nucl. Chem. (1981)
(C. C. Lin, General Electric Company)

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Figure 2. Partition of Iodine between Water and Gas Phase

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RHB 146-81

June 25, 1981

Dr. Richard S. Denning
Senior Research Leader
Nuclear Systems Section
Battelle Columbus Laboratories
505 King Avenue
Columbus, OH 43201

Dear Dr. Denning:

SUBJECT: LIAISON ON DRAFT NUREG-0772

We appreciated your visit to San Jose on April 8, 1981. It was a valuable meeting in that it enabled us to provide you information to clarify our technical positions on the draft of NUREG-0772 as discussed in your letter to me dated March 23, 1981. This letter and its Attachment #1 provide summary responses to each of the topics in your letter. Included as Attachment #2 is a copy of the viewgraphs used in the April 8, 1981 presentations. Also included is a copy of our recently completed information report NEDO-25240, "Suppression Pool Scrubbing Factors for Postulated Boiling Water Reactor Accident Conditions."

With reference to your letter of April 10, 1981 to Mel Silberberg, USNRC, we wish to reaffirm our prior statements that we are not only "willing".. but eager, "...to assist in providing data needed to assure that the plant is modeled accurately." We expect that you may need additional detailed information and discussion on the GE-BWR process system, its engineered safety features, and both the Mark I and Mark III multiple barrier containment systems. We look forward to supporting your efforts on BWR PRA analyses. Our overall objective is to insure that safety of the BWR is properly represented. This interest applies especially to the Mark III configuration with its drywell and suppression pool providing extra fission product retention barriers within the primary containment.

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If you have any further questions, K. W. Holtzclaw [(408) 925-2506] of my staff will continue to be available as your contact on either the subject report or BWR-PRA analyses.

Very truly yours,

RHBuchholz

R. H. Buchholz, Manager
BWR Systems Licensing
Nuclear Safety and Licensing Operation

RHB:csc:rf/162-G

Attachments

cc: M. Silberberg, USNRC
R. R. Sherry, USNRC

ATTACHMENT NO. 1

The contents of the attachment to the letter dated March 23, 1981 from Richard S. Denning, Battelle, to R. H. Buchholz, General Electric, concerning the first GE comment letter of March 17, 1981 to the USNRC on the working draft of NUREG-0772 have been divided into the following subjects. The GE clarifying comments are noted under each subject.

I. Containment Failure Assumptions

II. ECCS Pumping of Saturated Water and Alternate Water Source Availability

III. Chemical Form of Iodine

IV. Influence on Consequences of Iodine Chemical Form and Suppression Pool Removal

V. Consequences Have Not Been Overestimated by Orders of Magnitude

VI. Fission Product Transport Analyses

VII. GE Comments on Impact on Risk

JMS:csc:rf/162-I1

I. Containment Failure Assumptions

Denning Statements

For the two cases analyzed in which the pool is boiling, the containment is calculated to fail in the suppression pool area prior to fuel melting. Not only is there great uncertainty regarding the amount of retention for particulate and elemental iodine in a boiling pool but in these cases there is some question as to whether there would be water in the pool.

Consequences have not been overestimated by orders of magnitude. As discussed above in some accident sequences, the suppression pool would not only be boiling at the time of fission product release but, in addition, failure in the suppression pool structure would precede fuel melting.

As stated earlier, in those sequences where a DF of 1 was assumed, the containment had failed in the region of the wetwell prior to core meltdown. Thus, there are two questions involved: the effectiveness of a boiling pool in the removal of fission products and whether the release will be into water because of the location of the failure in the wetwell region. The time of containment failure prediced in the AEδ sequence is very sensitive to the amount of oxidation of zirconium occurring in the period before pressure vessel failure. In the analysis presented, virtually all of the zirconium was reacted following slumping into the lower plenum. The models describing behavior in the lower plenum are very uncertain. For analyses with less reaction of the zirconium, the failure time is extended.

GE Clarifications

Generalized statements cannot be made for all BWR containment types. These apparently are intended to apply primarily to the Mark I configuration (especially on assumed failure location); they should be reconsidered and reconstructed separately for Mark I and Mark III evaluations.

With regard to Mark I, in discussion in the April 8, 1981 meeting it was indicated that there was no specific justification for the assumed failure in the torus suppression pool; that really this was no change from the WASH-1400 assumption; no failure hole sizes were evaluated; and that Mark I treatment was probably conservative. Also the question arose as to whether reactor vessel motion could cause severance or damage to ECCS penetrations.

In Dr. Love's presentation, it was indicated that for Mark I, the licensing basis design pressures were about 62 psig for the drywell and 35 psig for the suppression pool torus. However, only two plants (Oyster Creek and Nine Mile Point 1) were built to the 35 psig design pressure. In order to facilitate primary containment pressure testing, all other Mark I plants have a torus design pressure of 62 psig also.

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We believe there is no basis for the assumption of failure in the torus, particularly below the torus water level, and no basis for the failure assumption at twice design pressure. A detailed structural analysis would be required. Failure modes would be plant specific. Failure will most likely be initiated in some constraining detail location, and such locations are to be found in the drywell structure as well as in the torus structure. In any case, failure would be by cracking with no catastrophic aperture expected. Dr. Love pointed out the limited cracking which occurred in the Kahl plant pressure test. That vessel was constructed to the less restrictive German codes, rather than the ASME codes. Dr. Strod advises that in a Mark I configuration the probability is small that containment deflection would cause severance or other damage to ECCS equipment.

With regard to Mark III, it was concluded that the early draft NUREG-0772 treatment was probably most conservative, and that Mark III containment failure location or severity had not as yet been analyzed specifically.

With regard to the assumption that a Mark III primary containment would fail at twice the design pressure of 15 psig and that the failure would cause a loss of the suppression pool water source, reference is made to J. E. Love's five viewgraphs on the subject. The timing and sequence of containment failure is crucial to proper evaluation of fission product location within the multiple barrier system. Dr. Love recommended that failure due to overpressure be assumed at a minimum of four times the design pressure, or 60 psig. Also, the GE studies strongly suggest that such failure would initiate by buckling and cracking in the torispherical dome region near the top of the building rather than in the suppression pool region. Our analyses are based in part on empirical information of British origin, and on the behavior of a torispherical shell.

An important conclusion is that the Mark III drywell structure within the primary containment (with yield stresses around 200 psig) would not be expected to suffer failure prior to the above postulated primary containment failure. It was noted that in NUREG-0772 there were no BWR scenarios which bypassed the Mark III suppression pool.

With regard to possible failure of the Mark III suppression pool, as outlined in D. M. Rastler's presentation outlined the substantial structures which protect pool integrity. Consider the extreme scenario as in the AEγ sequence of core material penetration of thimbles at the bottom of a BWR reactor vessel. The molten core-concrete interaction would occur in the pit below the general floor level of the drywell. The very substantial reinforced concrete structure around this central portion of the drywell floor area prevents any direct access of this molten core-concrete mixture to the region of the suppression pool. Additional features protecting the suppression pool water include the weir wall, the five foot thick drywell wall, the ten to fifteen foot thick floor under the pool and drywell, and the eight foot thick wall around the pool outer radius. This outer wall thickness is a "fix" which has been applied to most Mark III plants and in combination with the three foot thick shield building wall provides this eight foot thickness outside the primary containment steel to a height above the pool water level. In addition, in all Mark III plants, ground elevation is also above the pool water level.

Because of this significant protection of the suppression pool and because of the substantial nature of the drywell structure, GE has concluded that structural failure in the pool area leading to loss of pool water, and drywell and/or containment deflection leading to interference with ECCS functions either by severing of piping or by allowing a steam environment in the ECCS rooms would be very low probability occurrences. Least of all would be any expected physical interference with suppression pool cooling by the RHR system.

In the various BWR degraded core accident scenarios, those fission products not absorbed by the water prior to reaching the suppression pool surface would be available for transport in the Mark III primary containment building. In the event that the primary containment pressure continues to increase due to loss of decay heat removal or non-condensible gas generation, GE analyses indicate that catastrophic failure is not expected. As outlined above, buckling and cracking may occur near the top of the reactor building many hours after the start of the scenario. Therefore, fission products emerging from the suppression pool surface must travel for approximately 130 feet vertically through a pathway of stairways, floor gratings and compartments with substantial plateout surfaces before reaching the primary containment steel cracking locations. The venting through the cracks would be to the encompassing shield building with exhaust through the standby gas treatment system and/or directly through cracks or other leakage points. Significant areas for plateout and deposition are available in the relatively low temperature spaces of the shield building and other compartments of the secondary containment zone.

For these reasons, GE considers that realistic treatment of Mark III primary containment failure requires evaluation of the fission product transport and retention provided by the multiple containment configuration.

See also related issues under "ECCS Pumping of Saturated Water and Alternate Water Source Availability."

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II. ECCS Pumping of Saturated Water And Alternate Water Source Availability

Denning Statement

The ability of the pumps in the newer reactor coolant system designs to pump saturated water does not assure their survival in the depressurization transient following containment failure. We would be interested in any information your staff has regarding these conditions.

GE Clarifications

As noted in the transcript of the peer review meeting, the electrically driven ECCS pumps in all BWR/5 and BWR/6 plants, and in the BWR/4 plants not yet operating, are fully qualified to pump saturated water. Thus, we do not expect loss of electrically driven ECCS pumps during turbulent boiling.

In the various accident scenarios there is at least a three hour time delay between the time of primary containment cracking due to overpressure and the beginning of core melt due to loss of makeup water. It is expected that the period of turbulent boiling of the suppression pool during primary containment depressurization would be much shorter than this three hour period.

In addition, other water sources are available and other actions can be taken during this period. ECCS pumps can be turned off during the limited period of turbulent boiling. Both RCIC and HPCS or HPCI pump suction can be switched to the cool water in the condensate storage tank which has a dedicated reserve for this purpose.

The primary containment cracking occurs long after the accident scenario starts (more than about 30 hours later in a Mark I, and more than about 15 hours later in a Mark III). The control rod drive (CRD) water system obtains its water from the condensate storage tank and has a pumping capacity sufficient to compensate for core decay heat at these times.

The long elapsed time to primary containment cracking provides ample opportunity to provide backup water makeup to the condensate storage tank from dimineralized water storage, fire water, potable water, or other sources that may be available at a particular plant or multi-plant site.

See also related issues under "Containment Failure Assumptions."

III. Chemical Form of Iodine

Denning Statement

CsI or HI dominant form, but some I_2 in some situations. As discussed in Chapters 4 and 5, the results of investigations by ORNL and Sandia staff indicated that CsI would be the predominant form expected to be released to the drywell for most accident sequences. However, the data are inadequate to preclude the possibility of release as elemental iodine.

GE Clarifications

It was noted that the only scenario which had identified the formation of elemental iodine was the AB failure of the "hot leg" in a PWR. There was no BWR scenario identified in NUREG-0772 which projected the formation of elemental iodine. It is recommended that BWR and PWR scenarios be clearly differentiated, and that overall statements not be made unless warranted.

The study by D. M. Rastler on suppression pool scrubbing which was summarized in the April 18, 1981 meeting concludes that scrubbing of elemental iodine would be highly effective ($DF >> 10^2$) in a subcooled pool, and very effective ($DF>30$) in a saturated pool. Since the meeting, the report on this study (NEDO-25240) has been issued, and a copy is enclosed for information and reference.

IV. Influence on Consequences of Iodine Chemical Form and Suppression Pool Removal

Denning Statements

Iodine chemical form does not have a major influence on consequences. We agree that changes are required in the report to clarify the meaning and reasons for the apparent insensitivity of the calculated results to the chemical form of the iodine. The mechanisms controlling the removal of aerosols and elemental iodine in bubbles passing through a suppression pool are substantially different. In our investigation of the BWR sequences, we treated DF's parametrically to examine the potential importance of retention in the pool. This approach was taken because we did not have an experimentally verified model for treating removal in the suppression pool available to us. Recognizing that the DF of 100 used in the sequences analyzed with a subcooled pool could be low, we intend to also include an analysis of these sequences with a DF of 1000.

Since we did not have an empirically supported model to calculate the amount of decontamination in the pool, we treated the DFs parametrically to see the difference in results. In the final report, we will include results with DF = 1000 for the cases where the suppression pool is subcooled.

From your discussion, you appear to feel that cesium iodide would transport through the suppression pool as a vapor. Our analyses indicate that CsI would condense on particles and transport through the pool as a suspended aerosol in the rising bubbles. The CsI partition coefficient is not relevant to this calculation. Our feeling was that the data on aerosol removal in the suppression pool were inadequate to predict a DF. We agree that the mechanisms for removal of elemental iodine and particulate iodine differ substantially and that the DFs could be greatly different. We will try to make this clearer in the report.

GE Clarifications

The above statements refer to retention in a subcooled suppression pool. For a saturated pool, as detailed in the new GE report NEDO-25240, where the pool is "steaming" rather than undergoing turbulent boiling, very effective retention will still occur. It is suggested that retention in a turbulently boiling pool is not an issue. As summarized in the GE clarifications on subjects I, II, and III above, in the various accident scenarios fission product transport through th pool does not occur at the same time as the turbulently boiling period caused by the postulated primary containment cracking and venting.

With regard to the influence of chemical form on consequences, the report should recognize the differences between BWR's and the various PWR's with regard to the process systems and the various forms of containment systems for each reactor type. The report should not evaluate an accident scenario for one reactor type and one containment design, and then either state or imply by omission that the consequences apply to another reactor type, or even to a different containment configuration for the same reactor type. Features which impact directly on fission product behavior should be individually considered.

V. Consequences Have Not Been Overestimated by Orders of Magnitude

Denning Statement

As discussed above in some accident sequences, the suppression pool would not only be boiling at the time of fission product release but, in addition, failure in the suppression pool structure would precede fuel melting.

GE Clarifications

The original GE comment was that the NUREG-0772 working draft overestimated consequences by orders of magnitude in the worst accident scenarios, as the overall fission product retention factor between the core and the outside atmosphere was stated to be ten or less. This overall conclusion should be reevaluated based on the GE clarifications for subjects I, II, III, and IV above. Reevaluation should especially consider:

a) Magnitude and location of BWR primary containment cracking,

b) Timing of fission product transport with respect to (a),

c) Effective suppression pool scrubbing action in all BWR scenarios, and,

d) Fission product retention by plateout and other natural phenomena in the BWR compartmentalized containment systems.

Further overall fission product transport mitigation apparently will result from changes in Chapter 6 of NUREG-0772 recognizing fission product retention within the reactor system.

Based on these technical considerations, it is concluded that the NUREG-0772 draft Abstract did overestimate consequences by orders of magnitude.

VI. Fission Product Transport Analyses

Denning Statements

In Appendix A, discussion of accident sequences was divided into two sections: A.2.1 Conditions in the Reactor Coolant System and Drywell, [and] A.2.2 Containment Conditions. The intent was to consider the flow path and conditions in the reactor coolant system and drywell generically in A.2.1 and then in A.2.2 to consider the flow path and conditions in the suppression pool and the vapor space of the wetwell for the three different principal design variations in the containment. This apparently led to your confusion as to why on page A-13 the discussion stops in the drywell. We will try to make this clearer in the text.

Modeling and assumptions used for dominant severe accident sequences. The transport pathway in the analysis is as you described. As discussed previously, the consideration of the flowpath on page A-13 ended with the drywell because the topic of Section A.2.1 is "Conditions in the Reactor Coolant System and Drywell." The following Section A.2.2 describes conditions in the transport pathway through the suppression pool to the overlying vapor space. We will try to clarify this discussion in the report.

GE Clarification

GE appreciates the additional information with regard to the division of work among the various contractor technical groups, and the clarifications planned for inclusion in NUREG-0772.

VII. GE Comments on Impact on Risk

Denning Statement

We are not as optimistic as you about the decontamination available from the pool for some very low probability accident sequences. Recent results for in-vessel behavior indicate that for some important sequences there will be significant retention of aerosols in the reactor coolant system. This could very well affect our perspective of risk dominant accident sequences.

GE Clarifications

GE is pleased that recent work not shown in the working draft of NUREG-0772, foresees that agglomeration and retention within the reactor system will mitigate fission product transport.

The concerns on "low probability accident consequences" arise from assumptions, not supported by any structural analyses, that primary containment failure is catastrophic and is synonymous with suppression pool ineffectiveness. See GE clarification on "Containment Failure Assumptions" above. This also should influence perspective of risk dominant accident sequences.

During the meeting of April 8, 1981 it was clarified that there were no BWR accident scenarios in the draft of NUREG-0772 that bypassed the suppression pool, and that if the pool were functioning, the overall accident consequences were minor. Reference is made to the GE clarifications under "ECCS Pumping of Saturated Water and Alternate Water Source Availability" above, where it is pointed out that the time of containment venting through cracking and the time of fission product transport to the pool are widely separated. It is believed that this should allay concerns on pool boiling during the venting period and questions on its fission product scrubbing action during this period.

It was agreed that the Reactor Safety Study overestimated BWR fission product transport through the containment system and amount released to atmosphere.

It is unfortunate that the NUREG report does not include a consideration of the probability of the various accident scenarios and the severity of off-plant consequences. Risk is a proper integration of all scenarios and their probabilities. The Abstract and Summary chapter need qualification in this regard, so that the report will not produce misunderstanding by those who choose not to recognize the report's limited scope.

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GENERAL ELECTRIC

NUCLEAR POWER ~~C-105~~

SYSTEMS DIVISION

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MFN-187-81
JMS-118-81

October 9, 1981

U.S. Nuclear Regulatory Commission
Secretary of the Commission
Washington, D.C. 20555

Attention: Director, Office of Nuclear Regulatory Research

Gentlemen:

SUBJECT: DOCUMENT NUREG-0771 FOR COMMENT, "REGULATORY IMPACT OF NUCLEAR REACTOR ACCIDENT SOURCE TERM ASSUMPTIONS," JUNE 1981

This letter and its attached technical comments is in reply to the request for comment which appears on page i of the subject report.

Technical improvements to those aspects of the licensing process based on accident analysis are required now, not several years in the future. The subject document does not accomplish the re-examination recommended to the Commission in 1980. The extended delay in the schedule to even consider improvements is having an undesirable impact on the nuclear option and on current decisions on plant siting and emergency planning.

In the subject document, there appears to be no basic change in thinking since WASH-1400, while we believe that the companion technical document NUREG-0772 suggests many aspects not previously recognized of retention of fission products in the reactor system and in multi-barrier compartmented containment system. While we appreciate the Staff's desire to have such conclusions based on additional research work, we suggest that a technical redefinition for various accident scenarios done today would have far greater technical validity than the non-mechanistic assumptions of 1961 in TID-14844 which still form today's licensing basis. Reappraisal is required especially on the following subjects:

- NUREG-0771 defends the current licensing basis by holding that the worst-case lowest-probability scenarios are still likely.
- The document does not recognize that even in the serious scenarios there is considerable mitigation by virtue of the retention of most fission products in the reactor system and in a multi-barrier compartmented containment system.

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- o The stress on catastrophic containment failure is unwarranted for pressure suppression containment systems. Even if a containment were to lose its pressure integrity due to overpressure, it would most likely maintain its general configuration and retain its inner compartmentation. This is especially true for the BWR Mark III design.
- o The BWR Mark III containment system, the basis for many plants now in design and construction, isn't even mentioned, much less evaluated in the body of the report. Thus no technical recognition is given for the additional barrier -- the drywell and the suppression pool -- within the primary containment. This additional barrier and the pool scrubbing capability is expected to remain effective even if the primary containment was postulated to fail due to overpressure.
- o NUREG-0771 does not recognize that the BWR accident scenarios in its companion Technical Basis Report (NUREG-0772) all have reducing atmospheres so that elemental iodine formation would not occur.

We recommend that the subject "For Comment" document be revised to permit some of the needed advances in the licensing bases resulting from accident analyses to be made now. We will appreciate the opportunity for immediate review of BWR design basis accident assumptions. Mr. Joseph F. Quirk, Manager, BWR Systems Licensing (408) 925-2606 has responsibility for this subject.

Very truly yours,

Glenn Sherwood

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

Comments by the General Electric Company on NUREG-0771

Summary, P 2, ¶ 4

The wording of the last sentence in this paragraph could leave the uninformed reader with the impression that the majority of accidents which cause significant fuel damage result in the release of fission products from the core into an oxidizing environment therein resulting in negligible removal of released iodine. The authors of both NUREG-0771 and NUREG-0772, should recognize that for a BWR there are no design bases accidents nor any other accidents defined in Reg. Guide 1.70 which result in an oxidizing environment for those events which result in fuel damage which are of concern relative to the health and safety of the general public. This last sentence therefore needs to be rewritten or removed from this report.

Summary, P 5, ¶ 1

A reassessment of the design basis concept is in order, but the implication that Class 9 accidents should be considered in the design basis is unacceptable. The proper place to consider Class 9 accidents is in probabilistic risk assessments.

Summary, P 5, ¶ 3

The conclusion that the only way to introduce new source terms is via the various rulemaking processes will defer any technical revision of the design bases for an unacceptably long period. A great deal of reasonable revisions to Reg. Guides 1.3 and 1.4 can be done on the basis of information now available.

Section 2.1, ¶ 2 on P 7

Since members of the public, who are not familiar with the multiple assumed failures necessary to achieve a non-mitigated LOCA, may read this report it is suggested that the authors include a paragraph describing the extreme unlikelihood of a "guillotine" break LOCA and subsequent loss of all ECCS systems. Express this low probability event in terms understandable by the layman.

Section 2.2, ¶ 1 on P 7

The conclusion of catastrophic failure of the containment due to overpressure has no basis. For the BWR Mark I containment evaluated in WASH-1400, see Appendix VIII of the Reactor Safety Study, Pages 136-137. There it is stated that the overpressure is not expected to lead to structural failure of the containment as a whole. Current studies for the BWR Mark III containment reach similar conclusions. It is noted that nowhere in the body of report NUREG-0771 is the Mark III containment with its extra barrier - the drywell and suppression pool - inside the primary containment even mentioned much less evaluated.

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9/29/81

Section 3.1, ¶ 4 on P 11

With regard to making changes in the current accident source term assumptions, exception is taken to the statement that "a more accurate, or more realistic assumption would not necessarily result in an improved accident evaluation." As noted in their previous statements, the NRC has disregarded the 1% solid release, as being of radiological significance in comparison to iodine. It is presumed the staff choose to disregard the solids because they are less mobile, and more easily reduced by natural phenomena and filtration than elemental iodine. Therefore, if a more realistic assessment is made relative to iodine behavior the resulting analysis has to result in an improved accident evaluation.

Section 3.1.1, ¶ 1 on P 12

It is stated that the release of activity from an instrument line break accident is via an unfiltered path to the environment. This statement is not true for a BWR. All instrument lines which communicate directly with the RPV terminate either in the reactor building for a Mk I or Mk II containment or in the containment building for a Mk III. In any case each of these structures filter the activity as soon as the Standby Gas Treatment System (SGTS) is initiated. Only for the first 10 minutes following the accident is it assumed that release is directly to the environment. The significance of this 10 minute period must be related to the absence of significant core damage this early in an accident scenario.

Current NRC methods also neglect iodine retention at a water-steam separation, and assume that the activity in water is released to the atmosphere.

Section 3.1.1, ¶ 3 on P 12

It is implied that if CsI was released that filtration requirements would have to change to include HEPA filters. It should be noted that all BWR standby gas treatment systems include HEPA filtration already; therefore, no change would be required for a BWR.

Section 3.1.1, ¶ 5 and 6 on P 12

The statement that comparable releases of I_2 or CsI would occur for degraded core accidents is totally misleading. The behavior of CsI in accident scenarios will be totally different from that of I_2. CsI is more reactive than I_2 and would therefore not be available for release to the environment to the same degree as I_2; therefore, the release of CsI to the environment and its associated radiological consequences would be lower than I_2.

In addition, in a DBA with ECCS success the core is not degraded. If the core should be damaged, there is little iodine of any chemical form airborne for possible release as it would be retained in the suppression pool.

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Section 3.1.1, Table 3.1 on P 1

It is stated that "MSIV leakage limits" are the resulting reg. requirement for a Rod Drop Accident (BWR). Based on past licensing practice it is unclear how the authors arrive at this conclusion. The NRC does not give any credit (at least documented credit) for MSIV closure for a rod drop accident. The MSIV leakage limits have been based upon the LOCA.

It is stated that if CsI was released for a Fuel Handling Accident, that a new model and experimental data would be necessary for fuel pool scrubbing efficiency. This would be true if DF values greater than the 100 currently allowed were desired. However, since this accident is of no radiological importance (at least in a BWR) there would be no incentive to conduct such an experiment since the I_2 value of 100 is conservative relative to CsI.

Exception is taken to the conclusion that a somewhat larger exclusion distance would be required. There is no way to get 25% of the CsI airborne and thus available for leakage.

Section 3.1.1, ¶ 1 on P 14

The conclusion reached in this paragraph is based on the assumption that CsI will behave identically to I_2. Such an assumption is totally wrong for the majority (if not all) of the accidents one could reasonably envision for a LWR. Stratton's contention, as evidenced by TMI, that orders of magnitude reduction in iodine dose consequences are reasonable for the existing structure of DBA assumptions is correct and needs to be so recognized by the NRC staff.

Section 3.1.3, ¶ 3 on P 15

There is no apparent basis for the statement that "the fission products contained in this aerosol could have a substantial effect on environment qualification of equipment and shielding requirements" when the current basis is that 100% of the noble gases and 25% of the iodines are airborne with 25% of the iodines deposited on containment/structural surfaces and 50% of the iodine plus 1% of the solids deposited in the suppression pool (for a BWR).

Section 3.2, P 21

The conclusions of Section 3.2 are based on the assumption that dry airborne pathways exist from the degraded core through some uncomplicated containment space and then out of a containment wall which has failed catastrophically. Except for the noble gases, there are no accident scenario pathways which are not impeded by vast wet surfaces for plateout and pools of water to pass through. Even if the containment cracks due to overpressure causing even a rapid depressurization, the structure is still expected to remain intact, and any fission product group except the noble gases will be subject to significant retention.

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9/29/81

Section 4.2, last ¶ P 24

There is nothing in the TBR to suggest that the organic iodide which need be considered in accident scenarios is anywhere near the 4% level assumed in current regulatory guides.

Section 4.2, P 26 on R.G. 1.5

It is unrealistic to assume that coolant concentrations are the maximum permitted by technical specifications with no regard to actual experience. There is no basic relation between permissible and actual.

The assumption that the iodine in the coolant is released to the atmosphere in a steam line break neglects the fact that the water-steam separation is still occurring in the reactor vessel and the iodine is largely (~99%) retained in the water.

Section 4.3, P 30 (MSIV Leakage Control System) (LCS)

It is implied that charcoal filters are not currently part of the MSIV LCS. It should be noted that all BWRs which have a MSIV LCS also have charcoal filtration as part of the SGTS, to which the LCS effluent is routed.

Section 4.3 P 30 and 31 (Pressure Suppression Pool)

Due to the predicted cesium releases, the pH of the suppression pool during a degraded core scenario is calculated to be about 10 and thus the pool should effectively retain elemental iodine in the water.

For DBA type accidents (post-LOCA blowdown period) the flow rates during fission product release are relatively low, thus pool scrubbing should be as effective as in severe accident scenarios which also have relatively low flow rates.

It is unclear how the authors can reach the conclusion that for severe accidents the suppression pool would remove CsI with an efficiency comparable to that for I_2. Such a statement is not justified in this report nor in NUREG-0772. On the contrary, because of the greater solubility of CsI than I_2 (as stated on Page 2, Paragraph 4) it is expected that CsI would be removed by the suppression pool to a much greater degree than I_2.

Scrubbing of CsI should be much better than for I_2 in a condensing pool. CsI would hydrolyze rapidly into the water and remain while the fraction of unhydrolyzed I_2 would evolve from the water.

Section 4.3, P 31 (Standby Gas Treatment System)

The terminology used here indicates that reference is to a BWR Mark I containment, as the different features of the Mark III containment system are not mentioned.

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9/29/81

The references to the most severe accident sequences are based on the assumption of catastrophic failure rather than the more probable cracking and depressurization of the primary containment barrier.

The reference to failure of the reactor building walls should be clarified as applying to a Mark I secondary containment. In such a barrier system, the reactor building is designed to relieve pressure at panels high in the building above the principal reinforced concrete structure. While this route would bypass the SGTS, the release point would be elevated.

No mention is made of the secondary containment shield building in a Mark III standard plant. The extent and location of leakage following cracking of the primary containment would depend on the rate of primary containment pressure relief, and analysis of effects on the three foot thick reinforced concrete shield building and its interconnection with other spaces within the secondary containment barrier.

Section 4.3 2nd full ¶ on P 33

This is an important paragraph with which we can agree on the need for probabilistic risk assessment, and weighting the probability of each accident sequence to determine overall effectiveness of the safety features of a design to provide mitigation beyond their original design basis.

Section 4.4.1, Pages 33-37

The justification for the emergency planning distances of 10 and 50 miles appears to be based on current non-mechanistic source terms combined with theoretical high release fractions of the fission product groups to atmosphere such as were used in WASH-1400. Proper evaluation of fission product retention during degraded core conditions within the reactor system, natural mitigation phenomena within the containment barrier systems, and the extended time and duration of actual releases to atmosphere will cast a completely different light on the value of prompt and extensive evacuation with the attendant real hazards of such actions.

Section 4.4.2 ¶ 2 on P 38

This additional reference to "significant damage and potential rupture" of a primary containment barrier is also without technical basis.

Section 5 Conclusons, ¶ 4 on P 40

The analyses called for here almost certainly will result in lower source terms for several of the fission product groups, as current worst case assumptions (as in WASH-1400) include such high release fractions.

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9/29/81

Section 5, ¶ on ESF on P 41

The effectiveness of the suppression pool mentioned here is obviously for a Mark I containment design as no mention of the Mark III geometry is made in the body of report NUREG-0771. In the Mark III design the effectiveness of the suppression pool is enhanced by its protected location as an additional barrier within the primary containment.

Section 5, ¶ on Emergency Planning on P 41

The conclusion that emergency planning requirements need no modification is invalid as the conclusion is based on current non-mechanistic source terms and release fractions.

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9/29/81

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Sandia National Laboratories

Albuquerque New Mexico 87185

March 29, 1982

Dr. Lewis Hulman
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue, P802
Bethesda, M.D. 20767

Dear Dr. Hulman:

Enclosed are my comments regarding the report "Interim Source Term Assumptions for Emergency Planning and Equipment Qualifications." As we discussed via telephone on March 17, my expertise is in the systems analysis area, and thus I could only comment on Appendix A. Hope these are of some use.

Sincerely,

Greg J. Kolb

Greg J. Kolb
Nuclear Fuel Cycle Systems
  Safety Division 4412

GJK:4412:ep

Enclosure

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D.J. Kaib's Appendix A Comments

page A-6 — A station blackout results in a loss of reactor coolant pump seal cooling. A loss of seal cooling to Ⓦ pumps has been predicted to result in 300gpm LOCA/pump at 30 minutes (see ZION PRA). Blackout sequences may therefore lead to core melt even with the T-driven AFWS pump available.

page A-7 — The dominant contributors to the IS LOCA involved failure of one of the check valves to reseat, i.e. the leak failure mode. Failure to reseat is a "demand" failure occuring at time zero; yet the RSS used an hourly failure rate to predict this failure probability. There is no mathematical justification for doing this. One may be comparing "apples to oranges" if the risk of the IS LOCA is compared to the risk of other sequences.

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page A-8 – Battelle Columbus Laboratories (BCL) may have different thoughts on only assigning a ϵ to S_2D. More recent work (RSSMAP) suggests they would also assign a γ with a .5 probability and place it in RSS PWR release category 3.

page A-9, lines 5 & 13 – How could boiling occur *before* containment failure. Isn't the water in the pool subcooled relative to the containment pressure?

page A-12, line 10 – To my knowledge, BC– did not consider the effect of the quench tank in RSSMAP.

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page A-12, last sentence — Again, BCL may have different thoughts. See page A-5 comment.

page A-13 — the footnote should read "Hatch, Cybulskis, and Wooton (1981)"

page A-14 — the footnote should read "Kolb, et. al"

page A-14, line 19 — I question the statement "hydrogen burning would not be as likely..."! For Oconee, the γ failure mode was assigned a probability of .5 for the T_2MQ-FH. For Sequoyah it was 1.0. A factor of two difference in PRA is not that significant.

General comment — What was the criteria for accident sequence selection? It seems that important sequences were missed, e.g. the Oconee ATWS sequence -

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~~C-117~~

Sandia National Laboratories
Albuquerque, New Mexico 87185

April 7, 1982

RECEIVED
APR 13 1982

W. Pasedag
Accident Evaluation Branch
M/S P-802
USNRC
Washington, DC 20555

Dear Walt:

Re: NUREG/CR-2629, CRNL/TM-8274, and CRNL/TM-8275; "Interim Source Term Assumptions for Emergency Planning and Equipment Qualification"

As I discussed with you by telecon on April 1, I have not had the time to spend on these documents that they probably deserve. The response time required was very short, and I had conflicts with foreign travel, other pressing work assignments, and personal commitments. During that telecon you suggested I merely document my overview comments and not spend time in detailed review. This brief letter hopefully serves that purpose.

My comments are aimed towards equipment qualification concerns; as you know, we are involved with the USNRC/RES Program, Qualification Testing Evaluation (QTE), under Bill Farmer. I have also had several discussions with Susan Gull at CRNL during the writing of these reports.

--I find nothing particularly new in the reports. But that's not a surprise. I have, for years, argued that research needs to be done to advance the state-of-the-art. New, arbitrary assumptions (e.g., 50% core melt) are still arbitrary. We cannot advance the state-of-the-art by re-arranging the same old (largely non-applicable) data and words. The fission product release research at Sandia, Battelle, ORNL, etc., should be encouraged.

--I would recommend some of our QTE Program reports on fission product releases and "best-estimate" approaches.

--So while the lack of new information is obvious to everyone close to the subject, the report can still be useful if it can be used as argument to convince others of the need for research.

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--I remain concerned about the proliferation of seemingly disconnected USNRC safety-related equipment research projects. We need more cross-talk to avoid duplication of effort and reinvention of the wheel. I do recognize that a fresh look and independence can be useful at times.

I apologize for these minimal inputs, but perhaps you can use them. What is the plan for the reports? Will they be published or left unpublished?

Best regards,

Lloyd

Lloyd L. Bonzon
Systems Safety Information
Division 4445

LLB:4445:dcv

Copy to:
USNRC W. Farmer
ORNL S. Gull
4445 L. Cropp, File 5.4

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~~C-119~~

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Mr. L. G. Hulman
Chief, Accident Evaluation Branch
Division of Systems Integration, NRR
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

April 5, 1982

Dear Mr. Hulman:

Attached are comments on the Draft Report of NUREG/CR-2629 entitled "Interim Source Term Assumptions for Emergency Planning and Equipment Qualification". A number of comments relate to the discussion at the Peer Review Meeting held on March 18, 1982. Additional comments and recommendations on the broader issue of establishing an interim source term are also provided.

As indicated in the attached comments, it is recommended that an interim source term be issued in the short term (i.e., by June 30, 1982). This interim step should be limited to addressing the airborne radioiodine and waterborne cesium portions of the source term. Ample empirical evidence exists now to not only justify but require such adjustments to the source term at this time. The subject draft report does not add substantively to the source term information data base, and certainly doesn't constitute a technical basis for not moving forward with an interim source term in the short term.

I want to thank you for providing me the opportunity of attending the peer review meeting on this important report and hope that these written follow-up personal comments will be of some value in your deliberations as to how to proceed on the issue of an interim source term.

Very truly yours,

E. A. Warman

E. A. Warman
Chief Engineer
Nuclear Technology Division

Attachment

COMMENTS ON DRAFT REPORT - NUREG/CR-2629

"INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATION" AND ASSOCIATED PEER REVIEW MEETING ON MARCH 18, 1982

Prepared By
E. A. Warman
Stone & Webster Engineering Corp.

Scope of Comments

The following comments are not limited to the Draft Report of NUREG/CR-2629, hereafter referred to as the Draft Report. Comments of a more general nature are provided on the broader subject of establishing interim source terms. A recommended approach to be taken in establishing interim source terms is discussed herein.

Summary Comments on Draft Report

The report does not provide explicit recommendations for interim source term assumptions coupled with technical justifications for each recommendation. It does not propose an approach for estimating source terms. Nor does it develop criteria and assumptions to be used for estimating source terms for equipment qualification calculations and emergency planning.

What the report does do is essentially rehash the Reactor Safety Study (RSS) and Reactor Safety Study Methodology Application Program (RSSMAP) efforts in the source term area. This rehash is accompanied by some relatively minor improvements over the RSSMAP activities. There is absolutely no new ground broken in this study. It therefore adds very little to the solution of the interim source term issue.

A fundamental problem with this study is that it simply represents still another analysis with the MARCH and CORRAL computer programs. As was stated in the Peer Review Meeting on March 18, by Dr. W. Stratton ' "If you turn the same crank, you're going to get the same answers." Admittedly there were some changes in these analyses as compared with previous similar analyses.

However, these changes were not substantive and did not address the real issues which make all such analyses suspect (see later comments).

The old adage of "Garbage in - - garbage out" as applied to computer codes can be modified to read "Reasonable assumptions in - garbage out" if the processing of the input by the code results in unrealistic output no matter how good the input is. The MARCH code has been described by the ACRS and others as being so fundamentally flawed as to be virtually unuseable. The track record for the CORRAL code isn't much better. Therefore it should be kept in mind that a "new" analysis using these same codes is essentially no change at all from the RSS and RSSMAP studies. It is not surprising, therefore, that the results in this Draft Report agree well with the RSS. This close agreement is a direct measure of how inappropriate these answers are.

As indicated by Dr. R. Vogel of EPRI, at the Peer Review Meeting, the original MARCH code was found to contain 100 fundamental flaws. A recent meeting to review a more improved version of MARCH identified 43 fundamental flaws. As reported by Mendoza, etal., in the May 1981 issue of Nuclear Technology, the CORRAL code overpredicted the radioiodine released to the atmosphere at the SL1 accident by a factor of 215. Such codes do not provide "best estimate" results.

To rely on the use of such codes, as was done in this study, reflects a mindset that is apparently based on the thought process that argues that unless an analysis is performed with computer codes it is not an appropriate analysis - irrespective of how accurately the computer codes emulate reality. A series of hand calculations based on first principals and sound engineering judgement is far better than a lot of numbers from questionable computer codes.

The point to be made is that these codes do not adequately treat the most important physical, chemical and natural phenomena. In the case of MARCH, the code does not adequately treat the thermal-hydraulics. The draft report merely lists all of the important things that were not included. However it leaves the uninformed reader with the impression that these are merely uncertainties. The fact that virtually all of the ignored effects act to reduce the source term is not discussed.

General Comments

The Reactor Safety Study (RSS) as reported in WASH-1400 should not be viewed as the standard against which other source term investigations are measured. The value of the RSS is in the establishment of a baseline estimate of accident sequences and their associated probability of occurance. The consequence analyses in the RSS, including the source terms, should be recognized for what they are - that is overestimates in an attempt to define upper bounds of radiological consequences.

In judging whether a postulated source term is adequate - the measure should not be how it compares with the RSS or the more recent slight modifications of the RSS contained in the Reactor Safety Study Methodology Application Program (RSSMAP). The source terms used in those studies in themselves represent interim source terms for Probabilistic Risk Assessment (PRA). These are *defacto* interim source terms for PRA work. They have no direct applicability in establishing interim source terms for emergency planning and equipment qualification.

The establishment of interim source terms for emergency planning and equipment qualification should have as its foundation all of the available pertinent technical knowledge - published in the technical literature and made available to the NRC in peer review meetings and comment correspondence. The RSS and RSSMAP documents are only two parts of the mosaic.

Among the more important information which is useful in addressing the establishment of an interim source term is the experience gained from all previous reactor accidents, especially those involving complete or partial core melting. Those accidents, such as SL1 with little or not containment or in which containment barrier was breached should be awarded special attention and relatively high weight in the scientific judgement process. After all, these represent hard empirical observations. Also of great importance are the many large and small scale experiments.

In assessing the information from these and other sources, particular attention must be given to the physical and chemical characteristics of the released fission products, the time at which they were available for escape to the environs, the natural phenomena which were operative in each instance, and the effects of engineered safety features (including containment).

Above all else, the evaluation of these data must be realistic. There is no excuse for applying what is thought to be conservative judgements in assessing the scientific merit of these data. The entire source term issue has been plagued with the use of the word "conservative" for what in fact was "wrong" judgement. The place for conservatism is at the end of the scientific judgement process - where a "safety factor" should be applied to cover reasonably expected uncertainties. This factor should not be some unsubstantiated "umbrella" which is based on convoluted logic such as a highly improbable dry accident scenario in a light water reactor plant. Rather, this "safety factor" should be based on best engineering judgement - a practice which is widely followed in other industries (such as the elevator industry in which public health and safety is assured by the use of such factors).

Before addressing a new interim source term, it would be instructive to address the question: "What are the scientific bases for the present source term?" Subsets of this question should address such issues as: "What are the scientific bases for the assumption that the radioiodine postulated to be airborne in the containment is primarily in the elemental (I_2) form? What are the bases for assuming the iodine consists of 91% elemtal, 4 % organic, and 5% particulate material? There are many other subsets to the above general question. However the public safety would not be well served by a protracted investigation of the answers to these questions. Simple straight forward answers, with all their attendant qualifications are in order. The regulatory staff should not be pre-occupied with the uncertainties to the extent that it precludes answering the questions. Clear headed scientific judgement should replace the mindset which is so concerned about uncertainties as to lose sight of the facts of the situation.

The four most important factors which influence the establishment of the source term are:

- The physical and chemical characteristics of the fission products which are postulated to be airborne in the containment building.
- The natural phenomena which are operative-independent of mitigating actions.
- The temporal effects as a function of time after initiation of the accident events.
- The effects of engineered safety features.

The paramount importance of radioiodine in assessing the radiological consequences of postulated accidents requires that a separate and distinct radioiodine portion of the source term be established. It should not be grouped with other radioisotopes. Nor should it be put forth as a surrogate for other radioisotopes.

Recommendations

A. Short Term (i.e., by June 30, 1982)

Enough is now known to permit establishment of an interim source term for emergency planning and equipment qualification by:

1) Reducing the present airborne radioiodine source term by at least a factor of 20.

2) Modifying the postulated chemical form of the airborne radioiodine to be more nearly consistent with the chemical and physical properties of airborne radioiodine observed in actual accidents and small and large scale experiments.

3) Increasing the amount of Cesium from 1% to 50% for waterborne sources used in equipment qualification calculations.

B. Long Term (i.e., by January 1, 1984)

Based on the intermediate results available from ongoing research and development efforts, establish a revised source term to be applicable after January 1, 1984, unless further ammended by the Commission at the conclusion of the ongoing research programs.

The approach that should be taken in establishing such an a revised source term is outlined below:

Step 1

Prepare a matrix style listing of the postulated accidents (e.g., the TMLB'δ sequence from WASH-1400). This listing should include a very brief description of the features of the postulated accidents which directly affect the source term.

Step 2

Prepare a matrix style listing of the most realistic assumptions of the fraction of the available fission product inventories which are postulated to escape from the core, the pressure vessel, the primary coolant system, and the containment. These fractions should be multiplicative. They should also be specific to individual fission products and fission product groups (e.g., Xenon, Krypton, Iodine, Cesium and groups of other fission products).

Step 3

Prepare a listing describing the postulated chemical and physical form of these fission products at each step in the process (i.e., in the core, the pressure vessel, the primary coolant system, and the containment).

Step 4

Provide the best estimate as to the time sequence for the transport of the specific fission products and fission product groups in going from the core through the pressure vessel, the primary coolant system, and through leakage paths in the containment.

Step 5

Provide the best estimate of the effects of engineered safety features (such as containment sprays in the case of a PWR and the suppression pool scrubbing in the case of a BWR).

Step 6

Combine all of the information from the above steps and postulate the most realistic source term for each class of accident.

Step 7

Only after completing step 6, should the questions of uncertainty and the conservatism be addressed. The range of uncertainties for each of the "bottom line" realistic source term components should be explicity established. This of course will require engineering and scientific judgement.

Step 8

Having established the most realistic source term components and the engineering and scientific judgement of their attendant uncertainties, the question of the probability of the scenarios should be addressed. In particular, a rough estimate should be made of the probability associated with the upper range of the uncertainties for each fission product or fission product group for each accident scenario. In a word, the upper range of uncertainty must have associated with it a probability that uncertainty would become a real situation.

Step 9

Armed with the information from steps 1 through 8 - safety factors could be developed. An appropriate safety factor should be developed for each fission product and fission product group for each class of accident. An overall safety factor approach should be avoided.

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~~C-127~~

OAK RIDGE NATIONAL LABORATORY
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NUCLEAR DIVISION

POST OFFICE BOX X
OAK RIDGE, TENNESSEE 37830

March 11, 1982

Mr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Dear Dr. Hulman:

SUBJECT: PEER REVIEW OF DRAFT REPORT ON "INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATIONS"

Your kind invitation to participate in the March 18 peer review of the draft report of the above "Interim Source Term" - document has been received and noted with the following reservations.

A brief review note on two parts of the draft has already been forwarded on February 18 to R. O. Chester and her collaborators in which we indicate strong reservations about the extra conservatism (beyond WASH-1400) which we found in the chapter on Fission Product Release. It is probable that this situation arises mainly from the significance attached to the more recent SASCHA data from Karlsruhe.

Since we are personally involved presently in obtaining new data on fission product release with a large scale factor over the German work, it is probably unwise for us to participate in this review, although we are beginning to derive new data confirming the main points made in our comments.

I am sure Ms. Chester will be glad to discuss this with you and perhaps in the near future we will have real numbers to share with you.

Sincerely,

GW Parker

G. W. Parker
Chemical Technology Division

GWP:clf

CC: R. O. Chester
T. S. Kress

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OAK RIDGE NATIONAL LABORATORY
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POST OFFICE BOX X
OAK RIDGE, TENNESSEE 37830

March 8, 1982

Dr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Hulman:

SUBJECT: PEER REVIEW OF DRAFT REPORT ON "INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATIONS"

Thank you for your invitation to participate in the subject peer review.

I have not yet received the document for review (although I have seen several sections). I will be happy to review the entire document, but cannot complete my review in time for your meeting on March 18, 1982. Moreover, due to previous commitments, it will not be possible for me to attend the peer review meeting as presently scheduled.

In view of the very short time period between the anticipated receipt of the document and the review meeting, I would suggest that the meeting be delayed. If you elect to do so, I am available during the week of April 4 or April 25.

Sincerely,

A P Malinauskas

A. P. Malinauskas, Head
Chemical Development Section
Chemical Technology Division

APM:jh

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~~G-129~~

Battelle
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505 King Avenue
Columbus, Ohio 43201
Telephone (614) 424-6424
Telex 24-5454

March 8, 1982

Mr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Interaction
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Hulman:

I am writing in regards to your invitation to participate in the peer review of the interim source term report. I regret that I will be out of the country for the next week and a half. I also have a commitment on the eighteenth. I have reviewed parts of the report at the request of Ms. Niemczyk previously and have provided comments to her. I have asked Mr. Cybulskis of BCL to provide you with some written comments in which we collaborated.

Please do not construe any comments that we make as criticism of Ms. Niemczyk and her co-workers. She has been given a very difficult assignment and has made a very sincere effort to respond. It is my opinion, however, that it is premature for NRR to endorse a new source term for emergency planning and equipment qualification. Significant improvements in our understanding of fission product deposition in the primary system can be expected within the next year or so. These improvements can be expected to lead to predictions of greater primary system retention than currently indicated. A less conservative treatment, however, cannot be justified at this time. A conservative treatment of source terms as proposed in the interim source term report will be seen as a step backward by industry and will probably require revision shortly as more information becomes available through on-going research. In light of the above, I believe that the proper conclusions to be drawn from the subject report at this time are:

(1) Significant potential for retention of radionuclides in the primary system may exist at least for some accident sequences.

(2) It is premature at this time to take credit for primary system retention in defining regulatory source terms.

(3) For the interim period, pending the results of on-going research, WASH-1400 release terms should be used for emergency response planning.

Sincerely,

Richard S Denning/erc

Richard S. Denning
Senior Research Leader
Nuclear Technology and
Physical Sciences Dept.

RSD:erc

xc: R. Sherry, NRC
M. Cunningham, NRC
W. Pasedag, NRC
S. Niemczyk, ORNL

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~~C-131~~

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439

Telephone 312/972-7533

March 10, 1982

Dr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of System Integration
Office of Nuclear Reactor Regulation
Washington, DC 20555

Dear Dr. Hulman:

Subject: "Peer Review of Draft Report on Interim Source Term Assumptions for Emergency Planning and Equipment Qualifications"

On March 9, 1982 I received your letter of March 2, 1982 announcing the peer review meeting of March 18, 1982. I am certainly interested in participating in this peer review, however, I have yet to receive the draft report. As I understand this is a fairly detailed report and therefore could require substantial time for adequate review. Is it possible to postpone the meeting so that once having received the report some 2-3 weeks would be alotted for review.

Sincerely,

Carl E. Johnson

C. E. Johnson
Chemical Engineering Division

CEJ:smb

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U.S. Department of Energy The University of Chicago Argonne Universities Association

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March 9, 1982

Dr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Hulman:

In response to your March 2, 1982 invitation to participate in the peer review of the draft report on "Interim Scource Term Assumptions for Emergency Planning and Equipment Qualifications," I regret that previous committments will not permit me to attend the March 18, 1982 meeting.

As you may be aware, we have had substantial contact with the ORNL staff during the preparation of the subject report, including providing them with the details of a series of MARCH and CORRAL calculations. We have earlier reviewed portions of the draft report; a copy of our comments resulting from this review is attached. I will be happy to review the complete report after its issuance. Since at this writing I have not received it, it is doubtful that I will have the opportunity to do so and submit comments prior to the scheduled peer review meeting.

Sincerely,

Peter Cybulskis

PC/jwm

Enclosure

cc: M. A. Cunningham, NRC
W. Pasedag, NRC
R. Sherry, NRC

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UNIVERSITY OF COLORADO, BOULDER
Department of Chemistry

March 11, 1982

Dr. L. G. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration, NRR
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Hulman:

Thank you very much for your letter of March 2 informing me of the forthcoming Peer Review of a document addressed to the technical basis for estimating fission products behavior during LWR accidents. Indeed, this is an area in which I have considerable interest and one on which I have done extensive research over the years. Unfortunately, with less than two weeks of notice about the forthcoming meeting, I am unable to rearrange my teaching schedule in order to attend the meeting.

I hope that you will keep me on your mailing list, and if you have future meetings on this subject will invite me to attend; however, it certainly would be helpful if one could have at least a month or so advanced notice concerning meetings.

Sincerely,

Will Castleman

A. W. Castleman, Jr.
Professor of Chemistry

AWC:mr
xc R. O. Chester, ORNL

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OAK RIDGE NATIONAL LABORATORY
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March 12, 1982

Dr. L. C. Hulman, Chief
Accident Evaluation Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Hulman:

This letter is in response to your invitation to me to serve on a technical review committee for the draft report, "Interim Source Term Assumptions for Emergency Planning and Equipment Qualifications." Although the review meeting is scheduled for March 18, part of the report has not been received as of this date, March 12. Because of conflicting commitments I will be unable to participate in the review on March 18, nor will I have time to give the report the attention it requires prior to that date. If a more realistic schedule can be established, I would be happy to participate.

Sincerely,

D.O. Campbell

D. O. Campbell
Chemical Development Section
Chemical Technology Division

DOC:jh

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ENCLOSURE 4

PEER REVIEW COMMENTS ON "INTERIM SOURCE TERM ASSUMPTIONS FOR OPERATING REACTORS"

A. General Comments	Response Category*
1. Several people commented that the review was rushed.	0
2. Several people were not clear about the purpose of the study.	0
3. Several people commented on a lack of description of calculations through all the steps during an accident.	2
4. Several reviewers felt that there were excessive conservatisms in all the steps of calculations, and they questioned the credibility of numerical results.	2
5. Several people stated that a lot of research is going on and there will be a break through in the near future which will substantially improve the quality of source term calculation. Pending the results of this research, quantitative revisions of source terms are premature.	2
6. Several reviewers pointed out that there were a large number of problems with the MARCH Code which was the basis of all release calculations. It was suggested that the ANCHAR Code use should be investigated, and other codes surveyed.	1
7. The discussion of the limitations, gaps, and uncertainties should be expanded and their potential impacts noted.	3
8. Synergistics could increase or decrease the effects of a transport process. They should be discussed further.	3
9.. In the absence of good technology, rapid generalizations based on slender evidence should be avoided.	1

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*Response category "0" means that ORNL will respond by 3/31/82

Response category "1" means that ORNL can respond by 4/30/82, provided that the contract is extended

Response category "2" means that ORNL can respond by 6/30/82, provided that the contract is extended

Response category "3" means that a long-term research project beyond the scope of this study is required to resolve this issue.

10. The Zion/Indian Point studies should be looked at for handling uncertainties. 2

11. Radioactive decay should be considered in source term calculation. Chemistry of all species should be considered. 1

12. One commenter suggested that all the species except iodine and cesium should be ignored in source term calculations. 0

13. Difference of opinion on the chemistry problems should be stated and reflected in the results. One method suggested to accomplish this is to give a range of values indicating upper and lower bounds. 2

14. Less reliance should be placed on unpublished information not publicly available. 1

15. Documentation should include provision of definitive settings of Emergency Action Levels. 0

16. Relevent research work should be discussed. 0

17. The work should be continued. A continuous peer review committee should be set up to guide the writing of the final report. 0

B. Comments on Accident Sequence Selection

1. What was the basis for selection of accident sequences for the study? 1

2. Event "V" should not be considered since post-TMI modifications have reduced the probability of this accident. 1

3. Steam explosions should not be considered. 0

C. Comments on Releases From the Fuel	Response Category
1. The temperature transient calculations of the MARCH Code were questioned with respect to their applicability to fission product release rate correlations.	1
2. Is the 0772 method applicable to releases from fuel after slumping (Including changes in S/V ratios)	1
3. Has the decay heat depletion due to fission product release from fuel been considered in calculations?	1
4. Is there a consistency in treatment of fission product aerosols and structural material aerosols?	0
5. The calculation of releases from the fuel did not consider the rationale in the RSS. The 0772 estimates were no more valid than those of the RSS as far as Te, Ba, and Sr releases were concerned.	1
6. It was commented that SASCHA data is not suitable for determining absolute release rates because of intrinsic limitations of its expreimental set-up.	1
7. Was the solubility of lanthanum considered in leach release from the fuel?	1

D. Comments on the Transport Through the Primary System	Response Category
1. Several people wanted to know the details of the primary coolant retention calculations.	2
2. Retention of aerosols in the primary system was over emphasized. And the retention of the material on internal surfaces was under emphasized	1
3. The chemistry of tellurim in the primary system should be considered.	2
4. The length of the aerosols residence time is of no importance in the primary coolant system becasue aerosol concentrations are always very high.	1
5. Resuspended aerosols would be too heavy to be transported out.	1
6. Reactions of iodine in the primary coolant system were not considered.	1
7. The estimates of aerosol retention in PCS in 0772 are underestimates for many reasons and not overestimates as suggested in the report.	1
8. Steam flow rates through the primary coolant system are always low except prior to any fission product evolution and after slumping of the core. During the critical fission product release period the steam flow rate would be low.	1
9. Delayed operation of ECCS would quench and attenuate aerosols.	1
10. Scrubbing of aerosols and vapors in primary system is inadequately treated.	1

11. Numerical values for retention in the primary coolant system are more precise than current state of knowledge. Quantitative estimates should be limited to orders of magnitude.	1
12. Resolving the uncertainties would tend to reduce the escape fractions	1

D. Comments on Transport through the Containment

	Response Category
1. Some people stated that CORRAL Code was not the state-of-the-art, 0772 would be better.	1
2. Some felt that the effect of presence of large quantities of water in the containment was not properly modeled.	1
3. Steam environment in the containment should be better modeled.	1
4. There is not enough analysis of the containment failure modes to pin down time when radionuclides would be released to atmosphere. The effects of scaling in containment failure analysis should be considered.	1
5. Many problems are associated with the use of CORRAL code.	0
6. Clogging of containment leakage paths was not considered	1
7. The NAUA results appear to be misplotted.	1
8. The report says that LMFBR codes are not applicable to LWR accidents.	1
9. Adjusting CORRAL results to account for acknowledged problems is technically indefensible.	1
10. It is stated that "because of the somewhat orbitrary nature of the definition and choice of the accident sequence ---" "--- problems with many of the detailed models --- are not overwhelmingly important". This statement is not defensible.	1

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(11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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INTERIM SOURCE TERM ASSUMPTIONS FOR EMERGENCY PLANNING AND EQUIPMENT QUALIFICATION

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
S. J. Niemczyk, and others

5. DATE REPORT COMPLETED
MONTH April
YEAR 1982

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

DATE REPORT ISSUED
MONTH June
YEAR 1982

6. *(Leave blank)*

8. *(Leave blank)*

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Division of Systems Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
B0767

13. TYPE OF REPORT
Technical

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9/5/81-4/30/82

15. SUPPLEMENTARY NOTES

14. *(Leave blank)*

16. ABSTRACT *(200 words or less)* The source terms recommended in the current regulatory guidance for considerations of light water reactor (LWR) accidents were developed a number of years ago when understanding of many of the phenomena pertinent to source term estimation was not well developed. The purpose of the work presented here was to review the literature on accident source term research and utilize the recent research to develop more realistic assumptions for calculation of accident source terms which could be used for regulatory purposes for two specific considerations, namely, equipment qualification and emergency planning.

The emphasis of this work was on developing appropriate assumptions for estimating the magnitudes of the radionuclide releases for various groups of accidents in each of the accident spectra of concern. The overall approach taken was to adopt basic assumptions and models previously proposed for various aspects of source term estimation and to modify those assumptions and models to reflect recently gained insights into, and data describing the release and transport of radionuclides during and after light water reactor accidents. The report presents results of sample calculations of accident source terms and compares the results with the other published results. The report also presents peer review comments on this study.

17. KEY WORDS AND DOCUMENT ANALYSIS
Nuclear Reactor Accident Source Term
Equipment Qualification
Emergency Planning
Interim Source Term Assumptions

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

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